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Revision 0

August 2004

Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs

Westinghouse

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Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs

August 2004

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WCAP-16072-P

Nuclear Regulatory Commission

Final Safety Evaluation

dated May 6, 2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 6, 2004

Mr. James A. Gresham, Manager
Regulatory and Licensing Engineering
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16072-P, REVISION 00, "IMPLEMENTATION OF ZIRCONIUM DIBORIDE BURNABLE ABSORBER COATINGS IN CE NUCLEAR POWER FUEL ASSEMBLY DESIGNS" (TAC NO. MB8721)

Dear Mr. Gresham:

On April 25, 2003, as supplemented by letters dated September 10, November 3, and December 5, 2003, and February 3, 2004, Westinghouse Electric Company (Westinghouse) submitted Topical Report (TR) WCAP-16072-P, Revision 00, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," to the staff for review. On March 31, 2004, an NRC draft safety evaluation (SE) regarding our approval of WCAP-16702-P, Revision 00, was provided for your review and comments. By letter dated April 8, 2004, Westinghouse commented on the draft SE. The staff's disposition of Westinghouse's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The staff has found that WCAP-16702-P, Revision 00, is acceptable for referencing in licensing applications for CE Nuclear Power designed pressurized water reactors to the extent specified and under the limitations delineated in the report and in the enclosed SE. The SE defines the basis for acceptance of the report.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish an accepted version of this TR, including a non-proprietary version, within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in appendices historical review information, such as questions and accepted responses, draft SE comments, and original report pages that were replaced. The accepted version shall include a "-A" (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions in this letter, that the TR is acceptable, is invalidated, Westinghouse and/or the licensees referencing the TR will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the TR without revision of the respective documentation.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director /RA/
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Safety Evaluation

cc w/encl:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16072-P, REVISION 00,

"IMPLEMENTATION OF ZIRCONIUM DIBORIDE BURNABLE ABSORBER COATINGS

IN CE NUCLEAR POWER FUEL ASSEMBLY DESIGNS"

WESTINGHOUSE ELECTRIC COMPANY

PROJECT NO. 700

1.0 INTRODUCTION

By letter dated April 25, 2003, as supplemented by letters dated September 10, November 3, and December 5, 2003, and February 3 and April 8, 2004, Westinghouse Electric Company (Westinghouse) requested review and approval of Topical Report (TR) WCAP-16072-P, Revision 00, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs." Zirconium diboride (ZrB_2) is coated onto the outer surface of the uranium dioxide (UO_2) fuel pellets prior to loading into the fuel rod cladding tubes rather than being mixed with the UO_2 directly as is done with other integral fuel burnable absorber (IFBA) materials. The large neutron absorption cross section of boron (B^{10}) holds down reactivity early in the cycle and permits longer full power operation. An advantage with ZrB_2 is that as the B^{10} neutron absorber depletes, no residual neutron absorber worth remains as is the case with erbium and gadolinium.

Westinghouse has considerable fabrication and operational experience with the ZrB_2 IFBA fuel designs within Westinghouse-designed pressurized-water reactors (PWRs). Approval of the TR would allow the ZrB_2 IFBA design in CE Nuclear Power (CE) 14x14 and 16x16 fuel assembly designs. In determining the acceptability of this TR, the staff reviewed four aspects of the ZrB_2 IFBA fuel implementation: (1) operating and fabricating experience, (2) fuel mechanical design, (3) safety analysis models and methods, and (4) design basis accident (DBA) radiological consequences.

2.0 REGULATORY EVALUATION

The use of ZrB_2 IFBA in Westinghouse fuel assembly designs was previously reviewed and approved as part of the VANTAGE5 fuel assembly TR, WCAP-10444, "Reference Core Report VANTAGE5 Fuel Assembly." Review of WCAP-16072-P focused on the potential impacts of extending this approved fuel design feature to CE 14x14 and 16x16 fuel assembly designs and their associated safety analysis methodologies.

Regulatory guidance for the review of fuel system designs and adherence to applicable General Design Criteria (GDC) is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design."

In addition to review of the fuel system design and associated safety analysis methodologies, this safety evaluation (SE) addresses the impact of the proposed fuel design change on fission product inventory and transport assumptions used in DBA radiological consequence analyses. These assumptions form part of the bases of the DBA radiological consequences analyses performed to demonstrate compliance with:

- accident dose guidelines in Title 10 of the *Code of Federal Regulations* (10 CFR) 100.11, "Determination of exclusion area, low population zone, and population center distance," as supplemented by accident-specific criteria in Section 15, "Accident Analysis," of the SRP,

- accident dose criteria in 10 CFR 50.67, "Accident source term," as supplemented in Regulatory Position 4.4 of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," and

- 10 CFR Part 50, Appendix A, GDC 19, "Control Room," as supplemented by Section 6.4, "Control Room Habitability System," of the SRP.

The current assumptions accepted by the staff, and to which the fission product inventory and transport for the proposed fuel design are to be compared, are provided in the regulatory guidance documents listed below. If there are no significant impacts on the previous assumptions, it can be reasonably determined that the prior analysis results continue to meet the regulatory requirements specified above.

- RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors"

- Safety Guide (SG) 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"

- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"

- RG 1.183

- RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors"

- SRP Section 15.0-1, "Radiological Consequence Analyses Using Alternative Source Term"

- SRP Section 15.3.3, "Reactor Coolant Pump Rotor Seizure"

- SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Appendix A

- SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," Appendix A and Appendix B

3.0 TECHNICAL EVALUATION

The ZrB_2 IFBA fuel rod design consists of a ZrB_2 coating on the outer diameter of UO_2 fuel pellets over the center axial region of the fuel stack along with cutback regions (i.e., regions without ZrB_2 coating) on both ends of the fuel rod. Lower U^{235} enrichment fuel pellets may also be used in a portion of the cutback region. The cutback regions may consist of solid, annular, or a solid and annular fuel pellet combination. In determining the acceptability of this TR, the staff reviewed four aspects of the ZrB_2 IFBA fuel implementation: (1) operating and fabricating experience, (2) fuel mechanical design, (3) safety analysis models and methods, and (4) DBA radiological consequences.

3.1 Operating and Fabricating Experience

Since the approval of ZrB_2 IFBA in Westinghouse fuel assembly designs as part of the VANTAGE5 fuel design review, Westinghouse has accrued more than fifteen years of fabricating and operating experience. In its September 10, 2003, letter, Westinghouse provided details of the fabrication history of IFBA fuel rods. Westinghouse has fabricated a significant number of ZrB_2 IFBA fuel rods and these rods have irradiation experience in over 40 commercial nuclear plants. This historical database includes variations in B^{10} loading and variations in cutback regions (both solid and annular pellets). Westinghouse states that no fuel failures have been attributed to ZrB_2 IFBA fuel rod design in the substantial operational history within the Westinghouse fleet and at a CE-designed PWR (Fort Calhoun).

Westinghouse's letters dated September 10 and November 3, 2003, also identified post-irradiation examinations of ZrB_2 IFBA fuel rods. The post-irradiation examinations revealed no profilometry anomalies in the coated fuel pellet region, no chemical interaction between the coating and fuel rod cladding, no incipient cracks in the cladding inner diameter, no excessive fuel pellet cracking, nor any anomalies in the fuel structure. The ZrB_2 coating effectively remains in place throughout the service life of the fuel.

The substantial fabrication and operational databases along with the post-irradiation examinations demonstrate the reliability of ZrB_2 IFBA fuel rods. Based upon review of Westinghouse's ZrB_2 IFBA fuel experience, the staff finds no reason to anticipate fuel reliability problems with the implementation of ZrB_2 IFBA in CE fuel assembly designs.

3.2 Fuel Mechanical Design

The implementation of ZrB_2 IFBA fuel in CE fuel assembly designs will not necessitate any physical design changes to the fuel assemblies (fuel rod, spacer grid, support plates, etc.) nor changes to their materials. The ZrB_2 coating will slightly increase the fuel pellet diameter. In addition, to compensate for the helium production associated with the B^{10} depletion, the IFBA fuel design may employ annular fuel pellets (to provide additional void volume) and the initial helium fill gas pressure may be adjusted.

SRP Section 4.2.II.A defines fuel system damage and fuel rod failure mechanisms. Of these phenomena, the following are potentially impacted by the implementation of the ZrB_2 IFBA design in CE fuel assembly designs.

Fuel Rod Internal Pressure

Due to the no-clad-lift-off (NCLO) maximum pressure criterion established in CE TR CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," the maximum predicted fuel rod internal pressures are constrained to prevent an outward clad creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length. Both ZrB_2 IFBA and non-IFBA fuel rods will continue to satisfy this fuel design limit.

Clad Stress

The NCLO pressure limit ensures that internal rod pressures are comparable between ZrB_2 IFBA and non-IFBA fuel rods. Since tensile cladding stresses are associated with internal fuel rod pressures, the tensile cladding stresses of the ZrB_2 IFBA fuel rods and the non-IFBA fuel rods will be comparable. Impacts of fill gas pressure on compressive cladding stresses are discussed below under cladding collapse.

Clad Strain

Westinghouse has evaluated the impact of rod internal pressure and the increased fuel pellet diameter for both the CE 14x14 and 16x16 fuel designs with the ZrB_2 IFBA fuel design. The approved FATES3B code is utilized to predict cladding strain as well as many other burnup dependent fuel performance parameters. The evaluations demonstrate that both fuel designs continue to satisfy the current cladding strain criteria.

Clad Fatigue

Westinghouse has evaluated the impact of rod internal pressure and the increased fuel pellet diameter for both the CE 14x14 and 16x16 fuel designs with the ZrB_2 IFBA fuel design. The approved FATES3B code is utilized to predict cladding strain during cyclic power maneuvers, core shutdowns, and anticipated operational occurrences (AOOs). The evaluations demonstrate that both fuel designs continue to satisfy the current cladding fatigue criteria.

Clad Collapse

Using approved methods including the CEPAN computer code, Westinghouse has evaluated the impact of rod internal pressure and the increased fuel pellet diameter for both the CE 14x14 and 16x16 fuel designs with the ZrB_2 IFBA fuel design. The evaluations demonstrate that both fuel designs continue to satisfy the current cladding collapse criteria in the active fuel region, The Westinghouse evaluation of cladding collapse in the plenum region of the rods demonstrates that cladding collapse would not occur if the radial support offered to the cladding by the plenum spring was factored into the calculation. The staff had a concern with credit for radial support offered by the plenum spring since this was a deviation from established methodology (e.g., CENPD-404-P-A, "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs"), and not part of the design basis for this component.

In its December 5, 2003, letter, Westinghouse stated that no indication of cladding collapse in the plenum region has been observed in their considerable operating experience with ZrB_2 IFBA fuel rods. In addition, Westinghouse provided the results of autoclave tests (at elevated temperatures and pressures) on a variety of fuel rod designs with both zircaloy-4 and ZIRLO™ clad material. These autoclave tests and supporting ovality measurements demonstrate that clad collapse is essentially terminated upon hard cladding-to-spring contact. Furthermore, Westinghouse states that future autoclave tests will be performed, when needed, to verify adequate plenum spring support for CE fuel designs. Based upon operating experience, supporting autoclave tests, and a commitment to validate adequate plenum spring support in future applications, the staff finds it acceptable to credit the plenum spring for cladding collapse evaluations in the plenum region.

Clad Oxidation and Hydriding

Clad reaction rates and the associated degree of oxidation and hydriding will not be significantly impacted by the implementation of ZrB_2 IFBA fuel designs. However, an increase in rod internal pressure has the potential to promote radially-oriented hydride precipitates during plant cool down. In its February 3, 2004 letter, Westinghouse stated that the tensile stresses and peak temperatures for operation at NCLO conditions were concluded to be well below the magnitudes that might result in adverse hydride reorientation. In their response, Westinghouse also stated:

> It is the intention to address adverse hydride reorientation for conditions where the plant will recover and restart (Condition I & II). It is not intended to address reorientation for events where restart is not possible without further evaluation of fuel system damage (Condition III & IV events).

Since clad hydride reorientation was not addressed for these events, the staff has instituted a condition requiring that this issue be evaluated prior to restart following a Condition III or IV event.

Pellet/Cladding Interaction

Current criteria on cladding strain and fuel melting will continue to apply to ZrB_2 IFBA fuel design. Furthermore, Westinghouse has demonstrated via post-irradiation examinations of ZrB_2 IFBA fuel irradiated at the BR-3 reactor and the NRU reactor that no chemical reaction occurs between the ZrB_2 coating and its transmutation products and the cladding and that there is no adverse impact on the performance of the fuel rod.

Fuel Rod Ballooning and Bursting

During normal operations, the NCLO maximum pressure criterion (established in CEN-372-P-A) constrains fuel rod internal pressures to prevent an outward clad creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length. Both ZrB_2 IFBA and non-ZrB_2 fuel rods will continue to satisfy this fuel design limit.

During AOOs and postulated transients, fuel rods with elevated clad temperatures may experience outward clad creep even below the NCLO criteria. This phenomena raises concerns about excessive rod ballooning affecting neighboring fuel rods and even rod bursting.

The staff had concerns with the surge in rod internal pressure exhibited by the ZrB_2 fuel rods (depicted in Figures 4.2-3 and 4.2-4 of the TR) which increases the likelihood that rod internal pressure would exceed system pressure during the first operating cycle. In its September 10 and November 3, 2003, and February 3, 2004, letters, Westinghouse stated that predicted rod internal pressures were higher than expected due to the conservative models. However, Westinghouse would not commit to ensuring that rod internal pressures remained less than system pressures during the rod's first cycle.

With the rapid build-up of rod internal pressure associated with ZrB_2, the likelihood of a single fuel rod exhibiting rod internal pressure (in excess of system pressure) concurrent with a rod power close to the peak pin is significantly increased. As a result, the probability of a fuel rod with rod internal pressure in excess of system pressure experiencing departure from nucleate boiling (DNB) induced elevated clad temperatures (during Condition III or IV non-loss-of-coolant accident (LOCA) events) is dramatically increased. The staff had concerns that the implementation of ZrB_2 would promote rod ballooning and even rod burst during these conditions.

In its September 10 and November 3, 2003, and February 3, 2004, letters, Westinghouse stated that clad burst was an acceptable mechanism and would be credited for terminating rod ballooning during ZrB_2 applications. In addition, Westinghouse also stated that an allowable rod burst philosophy was "implicitly recognized" in CEN-372-P-A. The staff does not agree with these assertions. Although fuel rod bursting is an acceptable phenomena explicitly recognized during lower probability LOCA and implicitly recognized during lower probability non-LOCA events (e.g., control element assembly ejection), the staff had concerns with extending fuel rod burst to all events that experience elevated clad temperatures. Furthermore, the staff had concerns that allowing clad burst would encourage the development of future clad materials which lack sufficient creep properties and reduce the defense-in-depth found in the existing licensing basis for the fission product barrier.

To avoid these issues, the staff has instituted a condition to preclude fuel clad burst during Condition I, II, and III events. For Condition IV non-LOCA events which predict clad burst, the potential impacts of fuel rod ballooning and bursting need to be specifically addressed with regard to coolable geometry, reactor coolant system (RCS) pressure, and radiological source term.

In conclusion, the staff recognizes fuel rod ballooning as a fuel coolability concern which must be addressed for all categories of events. Further, the staff recognizes fuel rod burst as a distinctive fuel failure mechanism which must also be addressed. Although both DNB-related clad failure and fuel rod burst involve elevated clad temperatures, the failure mechanisms are driven by different phenomena. As a result, fuel rod burst must be assessed independent of DNB-related clad failure.

Based upon review of the fuel system damage and fuel rod failure mechanisms, the staff finds the fuel mechanical design aspect of implementing the ZrB_2 IFBA design in CE fuel assembly designs acceptable subject to the limitations and conditions described in Section 4.0.

3.3 Safety Analysis Models and Methods

Changes in the fuel rod design introduced by the implementation of ZrB_2 IFBA design may include: (1) ZrB_2 coating on the fuel pellets in the central axial region of the fuel stack, (2) axial cutback regions with lower U^{235} enrichment, (3) axial cutback regions with annular pellets, and (4) an adjusted helium fill gas pressure. This section addresses the potential impact of these changes on safety analysis models and methods.

<u>Core Physics</u>

The neutron cross-sections and reaction rates of B^{10} have been modeled extensively with the currently approved Westinghouse core physics codes. PHOENIX-P and ANC are already licensed as the primary neutronic models for all Westinghouse reloads, most of which contain ZrB_2 IFBA fuel designs. Westinghouse has benchmarked DIT-ROCS to PHOENIX-ANC on plants containing erbia, gadolinia, and ZrB_2 IFBAs and has produced results that are essentially the same. Based upon Westinghouse's experience modeling boron and the equivalency of the computer codes, the staff finds the use of DIT-ROCS acceptable for the implementation of the ZrB_2 IFBA design in CE fuel assembly designs.

As a result of the rapid depletion of B^{10} in the ZrB_2 IFBA fuel design, peak soluble boron concentration may occur after beginning of cycle (BOC). As a consequence, peak positive moderator temperature coefficient (MTC) may occur later than BOC. Plant technical specifications (TS) surveillance requirements (SR) (e.g., Standard TS SRs 3.1.3.1 and 3.1.3.2) dictate MTC measurements to validate the physics predictions and ensure that plant operations remain within TS limits. The staff had concerns that current plant procedures for meeting these surveillance requirements may be inadequate based on an increasing trend in MTC at BOC.

In its December 5, 2003 letter, Westinghouse stated that they would recommend that the MTC SR be modified if several conditions existed. The staff believes that their concerns warrant an SE condition as opposed to a vendor's recommendation. As a result, the staff has instituted a condition requiring that licensees confirm that the peak positive hot full power (HFP) MTC is within the TS limits at the highest RCS soluble boron concentration predicted during full power operation. The peak positive HFP MTC shall be derived by adjusting the measured MTC at HFP BOC conditions to the maximum HFP soluble boron concentration expected during the cycle. Plant procedures used to perform MTC surveillance should be updated to reflect the calculated peak positive HFP MTC along with ZrB_2 IFBA's distinctive trend in RCS critical boron concentration.

Manufacturing tolerances associated with ZrB_2 IFBA fuel (e.g., B^{10} loading and axial cutback region variations) will impact detailed core physics predictions. In its September 10, 2003, letter, Westinghouse stated that these tolerances would be conservatively applied within local power peaking and stored energy calculations. The staff finds the application of these manufacturing tolerances acceptable.

Fuel Performance

In the TR and its September 10, 2003, letter, Westinghouse provided details of the FATES3B model and its application for ZrB_2 IFBA. The FATES3B models, including annular fuel pellets, have already been reviewed and approved by the staff. These models have been extensively benchmarked to experimental data, much of which contained annular fuel pellets.

Helium is generated as a result of the depletion of B^{10} in the ZrB_2 coating. Along with other fission gases, helium contributes to increased rod internal pressure. Updates to FATES3B for the implementation of ZrB_2 IFBA include B^{10} depletion and helium release equations. Westinghouse has benchmarked these new models to those already approved in the PAD fuel performance code and to detailed core physics depletions. The results show good agreement.

Based upon the information presented in the TR and in response to requests for additional information (RAIs), the staff finds the modified FATES3B models and their application acceptable for the implementation of the ZrB_2 IFBA design in CE fuel assembly designs.

Safety Analysis

For emergency core cooling system (ECCS) performance analyses, ZrB_2 IFBA fuel is represented via normal code input. ZrB_2 IFBA fuel characteristics are also input through interfaces with core physics and fuel performance models. However, in the ECCS performance models, solid fuel pellet models will be used to represent annular fuel pellets in the axial cutback regions. Westinghouse provided the results in the TR of demonstration analyses which establish that this modeling approach yielded conservative peak clad temperatures (PCT) and maximum cladding oxidation results. Based upon the conservative results, the staff finds this modeling approach acceptable.

Large break LOCA and small break LOCA demonstration analyses reported by Westinghouse reveal that aspects of ZrB_2 IFBA fuel designs, especially the impacts of rod internal pressure, have the potential to produce significant changes in the calculated results. The staff inquired about the plant-specific implementation analyses which would be necessitated by ZrB_2 IFBA fuel designs. In its December 5, 2003, letter, Westinghouse stated that determination of whether a full-blown LOCA analysis was required would be made via the normal reload design process and that the acceptance criteria and reporting criteria in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," would be met. The staff finds this approach acceptable.

For non-LOCA safety analyses, the main challenge of the implementation of ZrB_2 IFBA fuel designs is the influence of the cutback regions on power distributions. Fuel pellets in the axial cutback regions at the top and bottom of the fuel stack will not be coated with ZrB_2 and may contain a lower U^{235} enrichment and consist of solid or annular pellets. Westinghouse evaluations of ZrB_2 IFBA fuel designs credit lower power peaking in these cutback regions such that these regions will never be limiting. As a result, plant-specific core design guidelines or cycle-specific calculations need to be used to verify that required power margins in the axial cutback regions are maintained within safety analysis limitations.

3.4 DBA Radiological Consequences

The staff review of the Westinghouse regulatory and technical evaluations contained in the TR revealed that they did not address the impact of the proposed fuel design change on assumptions used in DBA radiological consequences that relate to the inventory and transport of core fission products. Westinghouse responded in its September 10, 2003, letter to the staff's RAI specifically addressing this topic.

The staff's approach to this review was to establish that the changes proposed by Westinghouse would not adversely affect assumptions used in the DBA radiological consequences analyses. If this determination can be made, re-analysis of the affected events by applicants who reference this TR would not be necessary. The findings of this SE are based on the descriptions of the Westinghouse evaluations and other supporting information docketed by Westinghouse. During its review of the proposed fuel changes, the staff identified several possible impacts warranting evaluation and resolution.

Impact of the ZrB_2 Coating on the Source Term

The staff used the source term data provided in NUREG-1465, "Accident Source Terms Light-Water Nuclear Power Plants," for comparison rather than the earlier TID14844, "Calculation of Distance Factors for Power and Test Reactor Sites," since the NUREG-1465 data are more closely derived from observed phenomena than were the data in TID14844. The deterministic source terms in TID14844 are insensitive to the issues at hand. Numerous licensees have applied for and obtained authorization to use the source terms contained in NUREG-1465. The staff has determined that if the proposed changes can be shown to have minimal impact on the NUREG-1465 source term data then the same conclusion would also apply to the TID14844 data.

In response, Westinghouse states that the amount of fission products in the fuel rod gap is controlled by the temperatures of the inner regions of the pellets rather than the surface of the pellets. As such, the thin ZrB_2 coating is not expected to have a significant impact on the magnitude and mix of fission products in the fuel rod gap region given the relative cross-sectional dimensions of the fuel pellet and the coating. For accidents that progress beyond the release of gap activity, Westinghouse states that it is not credible that the ZrB_2 coating could significantly increase the magnitude and mix of fission products already projected to be released by NUREG-1465. The staff agrees that it is reasonable to assume that the ZrB_2 coating would not significantly effect the magnitude and mix of fission products and the timing of their release projected by NUREG-1465 for core melts associated with the early in-vessel release phase. The staff notes that the release assumptions provided in NUREG-1465 were based on sequences of severe accidents that involved substantial core damage. The mass of the added ZrB_2 coating is inconsequential in comparison to the mass of the other fuel and core constituents that would be involved in a core melt.

Westinghouse states that the mass of the ZrB_2 coating is small in comparison to the mass of metallic zirconium in the fuel pellets and fuel rod cladding. Also, the mass of cesium in the fuel matrix is on the order of ten times greater than the mass of radioiodine present. Thus, chemical reactions leading to cesium iodide are predominant. Westinghouse concludes that the added ZrB_2 will not affect the assumed chemical and physical form of released radioiodines. Westinghouse also stated that if the iodine were to combine with the slightly increased mass of zirconium to form zirconium iodide, there would not be significant impact on postulated doses since zirconium iodide, like cesium iodide, is an aerosol which is readily mitigated by natural processes and mitigation system operation. Based upon its consideration of the above information the staff has determined that there is reasonable assurance that the ZrB_2 coating will not impact existing source term assumptions.

Impact of Increased Helium Gas Pressure

With regard to the potential impact of increased helium gas pressure in the fuel pins on analysis assumptions regarding iodine scavenging by the spent fuel pool or reactor cavity, Westinghouse provided information that concludes that although there would be increased helium production in the fuel, it is not anticipated that the maximum internal fuel rod pressure for the ZrB_2 coated fuel would exceed the current design levels for CE plants. Westinghouse stated that the annular fuel pellets added to the fuel rods provide additional volume to contain the increased gas production. Also, cycle-specific core design constraints prevent current design pressures from being exceeded. In support of their conclusion, Westinghouse described an evaluation based on WCAP-7518-L, "Radiological Consequences of a Fuel Handling Accident," for fuel rod pressures of 1200 psig and 1500 psig. This evaluation determined that the iodine decontamination factors would be 580 and 473, respectively. The staff considered the methodology of WCAP-7518-L when it published SG 25. SG 25 provided a decontamination factor of 133 for fuel rod pressures up to 1200 psig. Westinghouse concluded that the factor provided in SG 25 would remain conservative. Based upon its consideration of the above information, the staff has determined that there is reasonable assurance that fuel rod design pressures of up to 1500 psig will not invalidate analysis assumptions related to iodine decontamination. The staff has also determined that this conclusion remains valid for the decontamination factor of 200 provided in RG 1.183 and RG 1.195, which supercede SG 25 for alternative source terms and TID14844 source terms, respectively.

Impact of the Annular Pellets on Fuel Gap Inventory

Westinghouse states that the potential impact of the annular pellets on fuel gap inventory will be small, as the fission product diffusion from within fuel grains and release from the grain boundaries will be the same for annular fuel pellets as for solid fuel pellets. Fission gas generation in the annular pellet would be proportionately larger at the same linear heat rate as for a solid pellet. The annular pellets constitute only about 10 percent of the active fuel length, typically the top and bottom 5 percent. In these regions of the core, the core power is lower and the linear heat generation rate is lower, resulting in lower pellet temperatures. Since the generation of fission products and the diffusion of fission products is proportional to temperature, there would be fewer fission products released to the fuel rod gap from the annular pellets. Westinghouse states that these differences were taken into account in the FATES3B fission gas analyses reported in the TR. Based upon its consideration of the above information, the staff has determined that there is reasonable assurance that the annular pellets will not significantly affect the fuel rod gap inventory.

Since these evaluations demonstrated that the changes did not have a significant impact on the DBA analysis assumptions, no dose calculations were necessary and none were performed.

4.0 CONDITIONS AND LIMITATIONS

Licensees referencing this TR to implement ZrB_2 IFBA in CE 14x14 and 16x16 fuel assembly designs must ensure compliance with the following conditions and limitations:

1. A license amendment is required to add this TR to the Core Operating Limits Report analytical methods listed in the licensee's TS.

2. Plant-specific core design guidelines or cycle-specific calculations shall be used to verify that required power margins in the axial cutback regions are maintained within safety analysis limitations.

3. Plant TS SRs on MTC validate the physics predictions and ensure that plant operations remain within allowable limits. In addition to current SRs, licensees shall confirm that the peak positive HFP MTC is within the TS limits at the highest RCS soluble boron concentration predicted during full power operation. The peak positive HFP MTC shall be derived by adjusting the measured MTC at HFP BOC conditions to the maximum HFP soluble boron concentration expected during the cycle. In order to ensure a conservative adjustment, a direct measurement of MTC is required at the highest RCS soluble boron concentration predicted during full power operation. This direct measurement is only required for the first application of ZrB_2 IFBA in a CE 14x14 or 16x16 fuel assembly design. During the first cycle implementation, Westinghouse shall provide the staff with a letter containing the following information:

    i. Measured HFP BOC MTC (TS SR),

    ii. Measured HFP MTC at highest RCS soluble boron concentration,

    iii. Calculated HFP MTC at highest RCS soluble boron concentration, and

    iv. Demonstrated accuracy of the calculated HFP MTC within current analytical uncertainties.

    In addition, plant procedures used to perform MTC surveillances shall be updated, where appropriate, to reflect the calculated peak positive HFP MTC along with ZrB_2 IFBA's distinctive trend in RCS critical boron concentration.

4. Prior to startup following a Condition III or IV event, licensees must evaluate clad hydriding to ensure that hydrides have not precipitated in the radial direction (in accordance with Section 3.2 of this SE).

5. CEN-372-P-A constraints and limitations with regard to rod internal pressure and DNB propagation must continue to be met. In addition, licensees must ensure that the following two conditions are satisfied:

   a. For Condition I (normal), Condition II (moderate frequency), and Condition III (infrequent) events, fuel cladding burst must be precluded for ZrB_2 IFBA fuel rods. Using models and methods approved for CE fuel designs, licensees must demonstrate that the total calculated stress remains below cladding burst stress at the cladding temperatures experienced during any potential Condition II or Condition III event. Within the confines of the plant's licensing basis, licensees must evaluate all Condition II events in combination with any credible, single active failure to ensure that fuel rod burst is precluded.

   b. For Condition IV non-LOCA events which predict clad burst, the potential impacts of fuel rod ballooning and bursting need to be specifically addressed with regard to coolable geometry, RCS pressure, and radiological source term.

5.0 CONCLUSION

The staff reviewed the effects of the proposed changes using the appropriate fuel design requirements of SRP Section 4.2 and 10 CFR Part 50, Appendix A, GDC and found that the TR provided reasonable assurance under both normal and accident conditions that CE fuel assembly designs implementing the ZrB_2 IFBA design would be able to safely operate and comply with NRC regulations.

The staff also reviewed the Westinghouse regulatory and technical evaluations related to the impact of the proposed fuel design change on the fission product inventory and transport assumptions used in DBA radiological consequence analyses. The staff finds the Westinghouse evaluations persuasive and supportive of the conclusion that the proposed fuel design will not significantly impact the fission product inventory and transport assumptions established in existing regulatory guidance and incorporated in current licensing basis analyses. The staff finds, with reasonable assurance, that should a design basis accident involving fuel of the proposed design occur, the radiological consequences will continue to comply with the applicable criteria identified in Section 2.0 of this SE. Therefore, the proposed fuel design is acceptable with regard to the radiological consequences of postulated design basis accidents.

Based upon its review of this TR, the staff finds WCAP-16072-P, Revision 00, acceptable. Licensees referencing this TR will need to comply with the conditions and limitations listed in Section 4.0 above.

Attachment: Resolution of Comments

Principal Contributors: S. LaVie
P. Clifford

Date: May 6, 2004

RESOLUTION OF COMMENTS

ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16072-P, REVISION 00,

"IMPLEMENTATION OF ZIRCONIUM DIBORIDE BURNABLE ABSORBER COATINGS

IN CE NUCLEAR POWER FUEL ASSEMBLY DESIGNS"

By letter dated April 8, 2004, Westinghouse provided comments on the draft safety evaluation (SE) for WCAP-16072-P, Revision 00, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs." The following is the staff's resolution of those comments.

1. Westinghouse Comment: Line 111, page 3, Section 3.1, "Operating and Fabricating Experience," first paragraph, last sentence – "Westinghouse claims that no ..."

   Westinghouse Proposed Resolution: "Westinghouse stated that no ..."

   NRC Action: The comment was fully adopted into the final SE.

2. Westinghouse Comment: Line 133-134, page 3, Section 3.2, "Fuel Mechanical Design," first paragraph – last sentence contained information that Westinghouse considered proprietary.

   Westinghouse Proposed Resolution: "... fuel design may employ annular fuel pellets (to provide void volume)."

   NRC Action: Last sentence now reads, "... and the initial helium fill gas pressure may be adjusted."

3. Westinghouse Comment: Line 147, page 4, Section 3.2, "Fuel Mechanical Design," "Fuel Rod Internal Pressure" – last sentence contained information that Westinghouse considered proprietary.

   Westinghouse Proposed Resolution: Delete last sentence.

   NRC Action: The comment was fully adopted into the final SE.

4. Westinghouse Comment: Lines 155-156, page 4, Section 3.2, "Fuel Mechanical Design," "Clad Stress" – last sentence contained information that Westinghouse considered proprietary.

   Westinghouse Proposed Resolution: "Impacts of a fill gas pressure on comprehensive cladding stresses are discussed below under cladding collapse."

   NRC Action: The comment was fully adopted into the final SE.

5. Westinghouse Comment: Line 168, page 4, Section 3.2, "Fuel Mechanical Design," "Clad Fatigue" – first sentence contained information that Westinghouse considered proprietary.

   Westinghouse Proposed Resolution: "Westinghouse has evaluated the impact of rod internal pressure and the increased fuel pellet diameter for both the 14x14 and 16x16 CE fuel designs with Zr B2 IFBA fuel design."

   NRC Action: The comment was adopted into the final SE.

6. Westinghouse Comment: Line 177, page 4, Section 3.2, "Fuel Mechanical Design," "Clad Collapse" – first sentence contained information that Westinghouse considered proprietary.

   Westinghouse Proposed Resolution: "... Westinghouse has evaluated the impact of rod internal pressure and the increased fuel pellet diameter for both ..."

   NRC Action: The comment was fully adopted into the final SE.

7. Westinghouse Comment: Lines 208-210, page 5, Section 3.2, "Fuel Mechanical Design," "Clad Oxidation and Hydriding" – last sentence reads "Since clad hydride reorientation was not addressed for these events, the staff has instituted a condition requiring that this issue be evaluated prior to restart following a Condition III or IV event."

   Westinghouse Proposed Resolution: Delete this sentence.

   NRC Action: The staff changed the text to clarify their position. Per telephone conferences on April 21 and April 22, 2004, the staff agreed to clarify the wording of lines 206-208 to read "Westinghouse also stated: It is the intention to address adverse hydride reorientation for conditions where the plant will recover and restart (Condition I & II). It is not intended to address reorientation for events where restart is not possible without further evaluation of fuel system damage (Condition III & IV events.)"

8. Westinghouse Comment: Lines 227-263, page 6, Section 3.2, "Fuel Mechanical Design," "Fuel Rod Ballooning and Bursting" – second, third, fourth, fifth and sixth paragraphs contained information that Westinghouse requested to be clarified.

   Westinghouse Proposed Resolution: Significant rewording of these paragraphs was proposed.

   NRC Action: Original paragraphs retained. To clarify the staff's position, a new paragraph was inserted between the original fifth and sixth paragraphs: "In conclusion, the staff recognizes fuel rod ballooning as a fuel coolability concern which must be addressed for all categories of events. Further, the staff recognizes fuel rod burst as a distinctive fuel failure mechanism which must also be addressed. Although both DNB-related clad failure and fuel rod burst involve elevated clad temperatures, the failure mechanisms are driven by different phenomena. As a result, fuel rod burst must be assessed independent of DNB-related clad failure."

9. Westinghouse Comment: Line 274, page 7, Section 3.3, "Safety Analysis Models and Methods" – Item 4 contained information that Westinghouse considered proprietary.

   Westinghouse Proposed Resolution: "... (4) initial helium fill gas pressure."

   NRC Action: The comment was adopted into the final SE as "... (4) an adjusted helium fill gas pressure."

10. Westinghouse Comment: Lines 297-301, page 7, Section 3.3, "Safety Analysis Models and Methods," "Core Physics" – second paragraph, last two sentences read "Further, until licensees have experienced several cycles of an increasing trend in RCS soluble boron concentration, a direct measurement of MTC is prudent. As a result, the staff has instituted a condition requiring that licensees confirm by direct measurement that the peak positive MTC is within the TS limits at the highest RCS soluble boron concentration predicted during Mode 1 operation."

    Westinghouse Proposed Resolution: Delete these two sentences or change the wording with a proposed rewrite.

    NRC Action: Per telephone conferences on April 21 and April 22, 2004, the staff agreed to clarify this paragraph with the wording as it appears in the final SE.

11. Westinghouse Comment: Lines 322-324, page 8, Section 3.3, "Safety Analysis Models and Methods," "Fuel Performance" – third paragraph contained information that Westinghouse considered proprietary.

    Westinghouse Proposed Resolution: Delete this paragraph.

    NRC Action: The comment was fully adopted into the final SE.

12. Westinghouse Comment: Lines 326-330, page 8, Section 3.3, "Safety Analysis Models and Methods," "Fuel Performance" – fourth paragraph contained information that Westinghouse considered proprietary.

    Westinghouse Proposed Resolution: Delete this paragraph.

    NRC Action: The comment was fully adopted into the final SE.

13. Westinghouse Comment: Lines 472-475, page 11, Section 4.0, "Conditions and Limitations" – third bullet reads, "Plant TSs SRs on MTC validate the pyhisics predictions and ensure that plant operations remain within allowable limits. In addition to current SRs, licensees shall confirm by direct measurement that the peak positive MTC is within the TS limits at the highest RCS soluble boron concentration predicted during Mode 1 operations."

Westinghouse Proposed Resolution: Delete this item or change the wording with a proposed rewrite.

NRC Action: Per telephone conferences on April 21 and April 22, 2004, the staff agreed to clarify this item with the wording as it appears in the final SE.

14. Westinghouse Comment: Lines 477-478, page 11, Section 4.0, "Conditions and Limitations" – fourth bullet reads "Prior to startup following a Condition III or IV event, licensees must evaluate clad hydriding to ensure that hydrides have not precipitated in the radial direction."

    Westinghouse Proposed Resolution: Delete this item.

    NRC Action: Original item retained, with reference to Section 3.2 of the SE.

15. Westinghouse Comment: Lines 480-481, page 11, Section 4.0, "Conditions and Limitations" – fifth bullet reads "CEN-372-P-A constraints and limitations with regard to rod internal pressure, hydride reorientation, and DNB propagation must continue to be met."

    Westinghouse Proposed Resolution: "CEN-372-P-A constraints and limitations with regard to rod internal pressure and DNB propagation must continue to be met."

    NRC Action: The comment was fully adopted into the final SE.

16. Westinghouse Comment: Lines 481-482, page 11, Section 4.0, "Conditions and Limitations" – fifth bullet reads "In addition, licensees must ensure that the following two conditions are satisfied:"

    Westinghouse Proposed Resolution: "In addition, when addressing DNB propagation, licensees must ensure that the following two conditions are satisfied:"

    NRC Action: Original wording retained.

17. Westinghouse Comment: Lines 484-494, page 11, Section 4.0, "Conditions and Limitations" – fifth bullet, Item a reads "For Condition I (normal), Condition II (moderate frequency), and Condition III (infrequent) events, fuel cladding burst must be precluded for all fuel types. Using current models and methods approved for CE fuel designs, licensees must demonstrate that the total calculated stress remains below cladding burst stress at the cladding temperatures experienced during any potential Condition II or Condition III event. To ensure that fuel rod burst is precluded, licensees must evaluate all Condition II events in combination with any credible, single active failure. The selection of limiting single failure shall include a loss of offsite power (LOAC). Unless the staff has previously approved a time delay for a LOAC following turbine trip for this category of event, the timing of the LOAC shall be coincident with reactor trip breakers open."

Westinghouse Proposed Resolution: "For Condition I (normal), Condition II (moderate frequency), and Condition III (infrequent) events, fuel cladding burst must be precluded for CE ZrB_2 rods using currently approved creep and rupture models approved for CE fuel designs."

NRC Action: The comment was partially adopted into the final SE. Per telephone conferences on April 21 and April 22, 2004, the staff agreed to clarify this item with the wording as it appears in the final SE.

18. Westinghouse Comment: Lines 496-498, page 11, Section 4.0, "Conditions and Limitations" – fifth bullet, Item b reads "For Condition IV non-LOCA events which predict clad burst, the potential impacts of fuel rod ballooning and bursting need to be specifically addressed with regard to coolable geometry, RCS pressure, and radiological source term."

Westinghouse Proposed Resolution: Delete this statement.

NRC Action: Original wording retained.

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List of Acronyms

BOL	Beginning-of-Life
CE	CE Nuclear Power
CEA	Control Element Assembly
DNBR	Departure from Nucleate Boiling Ratio
ECCS	Emergency Core Cooling System
EM	Evaluation Model
IFBA	Integral Fuel Burnable Absorber
LBLOCA	Large Break Loss-of-Coolant Accident
LHR	Linear Heat Rate
LOCA	Loss-of-Coolant Accident
MTC	Moderator Temperature Coefficient
NCLO	No-Clad Lift-Off
NRC	Nuclear Regulatory Commission
PCI	Pellet-Clad-Interaction
PCT	Peak Clad Temperature
PWR	Pressurized Water Reactors
SBLOCA	Small Break Loss-of-Coolant Accident
SER	Safety Evaluation Report
SIT	Safety Injection Tank
STP	Standard Temperature and Pressure
WABA	Wet Annular Burnable Absorber
Westinghouse	Westinghouse Electric Company, LLC
UO_2	Uranium Dioxide
ZrB_2	Zirconium Diboride

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Abstract

The Westinghouse Electric Company, LLC (Westinghouse) will introduce the zirconium diboride Integral Fuel Burnable Absorber (ZrB_2 IFBA) design into the CE Nuclear Power (CE) 14x14 and 16x16 fuel assembly designs. The ZrB_2 is coated onto the outer surface of the uranium dioxide (UO_2) fuel pellet stack prior to loading into the fuel rod cladding tubes rather than being mixed with the UO_2 as is done with other IFBA materials (e.g., erbia or gadolinia). As the B-10 absorber burns out, the fuel rod is left with no residual absorber worth as is the case with other IFBA materials like erbium or gadolinium. However, the burnout of the B-10 absorber results in production of helium gas which is released into the fuel rod plenum, [.] a,c The helium production effect on internal gas pressure and gas conductivity is taken into account in the design and safety evaluations in CE designed PWRs using the Nuclear Regulatory Commission (NRC) approved models and properties currently used in the Westinghouse designed PWRs. Neutronics codes already contain the capability to predict behavior of the ZrB_2 IFBA absorber. Consequently, only the simple addition of a ZrB_2 IFBA helium generation and release model in the FATES3B fuel performance code is required. Although FATES3B predicted fuel rod internal conditions (pressures, temperatures, etc.) are ZrB_2 IFBA specific for input to other analyses, no coding modifications are required for other design and safety analysis codes. It is the purpose of this topical report to describe the implementation and effect of using the ZrB_2 IFBA coating on the CE fuel assembly design and safety analyses.

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1.0 INTRODUCTION

1.1 PURPOSE AND SCOPE

Westinghouse Electric Company LLC (Westinghouse) customers operating CE designed pressurized water reactors (PWRs) have indicated a desire to implement zirconium diboride (ZrB_2) integral fuel burnable absorber (IFBA) fuel designs. Therefore, the ZrB_2 IFBA design is being introduced into the fleet of CE 14x14 and 16x16 fuel assembly designs. It is the purpose of this report to describe the implementation and influence of ZrB_2 IFBA on the CE fuel assembly design and safety analyses. Fuel performance, fuel mechanical design, Emergency Core Cooling System (ECCS) performance analyses for Loss-of-Coolant Accidents (LOCAs), non-LOCA transient analyses, and neutronics are described.

1.2 BACKGROUND

The Westinghouse Electric Company, LLC (Westinghouse) has had considerable fabrication and operational experience with the ZrB_2 Integral Fuel Burnable Absorber (IFBA). The ZrB_2 IFBA fuel has operated successfully for more than fifteen (15) years in a broad range of Westinghouse PWRs. ZrB_2 is applied as a very thin uniform coating on the outer surface of the UO_2 fuel pellet stack prior to loading into the fuel rod cladding tubes. As the B-10 absorber burns out, the fuel rod is left with no residual absorber worth as is the case with other absorber materials (e.g., erbia or gadolinia). However, the burnout of the B-10 absorber results in production of helium gas which is released into the fuel rod gas plenum. The neutronics effect, the helium production effect on internal gas pressure, and mechanical effect of the coating thickness are all taken into account in the design and safety evaluations for CE designed PWRs as described herein.

The ZrB_2 IFBA coatings may be natural or enriched with the B-10 isotope to increase the neutronic effectiveness. The enriched B-10 isotope is currently used in all Westinghouse IFBA designs. To obtain the proper peaking factor control, the ZrB_2 coating thickness is varied (i.e., 1.0X, 1.5X, 2.0X loadings, etc.). The ZrB_2 IFBA coating is applied over the center of the UO_2 pellet stack length and does not extend to either end of the fuel rod. The ends without ZrB_2 IFBA are referred to as cutback regions. The fuel pellets in the cutback regions may be solid, annular, or a combination of solid and annular geometry (i.e., solid pellet at the bottom of the pellet stack with annular pellets at the top of the pellet stack) and may be at reduced U-235 enrichment (blankets). However, the ZrB_2 IFBA coating is applied only to central solid fuel pellet stack. $ZrBr_2$ IFBA fuel rods are loaded into an assembly in specific core design locations as a matrix of ZrB_2 IFBA and UO_2 fuel rods. ZrB_2 IFBA fuel rods are introduced into the CE design, safety, and licensing analyses in a manner similar to that approved for Westinghouse designed PWR fuel assemblies (References 92, 94, and 95). Introduction of the IFBA design into CE designed PWRs requires a relatively small perturbation in CE design and licensing codes and methodology.

The B-10 isotope absorbs a neutron and fissions into helium and lithium. Helium is released from the thin coating into the fuel rod plenum by the time complete burnout is attained. This added helium contributes to the rod internal pressure at end of life. [

.] a,c This is typically referred to as the IFBA loading and is denoted as 1.0X, 1.5X, 2.0X, etc.

It is the purpose of this report to describe the implementation and effect of ZrB_2 IFBA on the CE fuel assembly design and safety analyses. Fuel performance, fuel mechanical design, ECCS analyses, non-LOCA accident analyses, and neutronics are described.

1.3 WESTINGHOUSE ZRB_2 IFBA EXPERIENCE

ZrB_2 IFBA fuel rods have been used successfully in Westinghouse designed PWRs for more than fifteen (15) years since the first region was loaded in 1987. Several hundred regions of ZrB_2 IFBA fuel have been used in more than forty (40) plants. In addition, Westinghouse had introduced the ZrB_2 IFBA fuel design in Fort Calhoun, a CE designed PWR, and ZrB_2 IFBA fuel was used in Fort Calhoun for several reloads. No fuel failures are associated with ZrB_2 IFBA coatings in Westinghouse or CE designed PWRs.

Current Westinghouse ZrB_2 IFBA fuel rod production is on the order of [ ] a,c rods per year. ZrB_2 IFBA fuel rods are used extensively in 14x14, 15x15, 16x16, and 17x17 Westinghouse PWR core designs, providing significant and sufficient experience to justify the introduction of the ZrB_2 IFBA fuel into the CE designed PWRs on a full batch basis. Westinghouse fuel rod designs, where ZrB_2 IFBA coatings have been used, range from [

.] a,c Post-irradiation examinations of ZrB_2 IFBA test rods revealed no profilometry anomalies in the coated fuel pellet zone, no chemical interaction between the coating and fuel rod cladding, no incipient cracks in the cladding inner diameter, no excessive fuel pellet cracking, nor any anomalies in the fuel structure. The ZrB_2 coating effectively remains in place throughout the irradiation.

1.4 SUMMARY

Helium gas generation and release models for the ZrB_2 IFBA coating have been incorporated into the FATES3B fuel performance code. Existing neutronics codes already contain the necessary models for ZrB_2 IFBA. The effect of ZrB_2 IFBA on mechanical design and safety analyses was evaluated. It is concluded that the influence of ZrB_2 IFBA is relatively minor and no significant design or licensing issues exist because of the introduction of the ZrB_2 IFBA design into CE designed PWRs.

Figure 1-1
Typical Fuel Rod Design
14x14 ZrB_2 IFBA

a, c

Figure 1-2
Typical Fuel Rod Design
16x16 ZrB_2 IFBA

a, c

2.0 ZRB$_2$ IFBA PROPERTIES IN DESIGN AND LICENSING

No new isotopic materials are being added to the ZrB_2 IFBA fuel rod. Neutronic properties of ZrB_2 are standard properties already existing in the Westinghouse neutronics codes for both Westinghouse and CE designed PWRs. Verification of the application of CE neutronics codes for the ZrB_2 design is provided in Section 3.1.

The addition of the ZrB_2 IFBA coating does, however, provide a helium source as the B-10 burns out. The helium is effectively accounted for in the FATES3B fuel performance code in much the same way as standard xenon and krypton fission products are tracked and taken into account.

In addition, the ZrB_2 coating effectively reduces the fuel-clad gap and affects pellet-clad mechanical interaction. The reduction in the as-fabricated gap and its effect on design and licensing are described below.

2.1 BORON DEPLETION CORRELATION

The fractional B-10 depletion from the ZrB_2 IFBA coating has been found to correlate well to fuel burnup and U-235 enrichment. Westinghouse developed a depletion correlation based on detailed physics analyses. The FATES3B depletion equation is identical to that used in the Westinghouse PAD fuel performance code, Reference 95, and is given by

[ ]a, c (1)

where [ ]a, c

This equation covers

- enrichments from 0.74 to 5.0 w/o,
- burnups from beginning-of-life to end-of-life, and
- is applicable to a broad range of assembly lattice types

The above conditions bound CE fuel designs.

[ ]a,c

2.2 HELIUM RELEASE

Absorption of a neutron by the B-10 isotope in the ZrB_2 (depletion) results in the production of one helium atom (He-4) and one lithium atom (Li-7). Thus, considering the mass balance from the nuclear reaction, the depletion of a lb-mole of B-10 results in a lb-mole of helium gas, and the balance remains as solid lithium. The gaseous helium escapes from the ZrB_2 IFBA coating and will contribute to the gas composition mix within the fuel-clad gap and other internal void volumes. Consequently, the helium contributes to fuel-clad gap conductance and fuel rod internal gas pressure. This helium is taken into account in the FATES3B fuel performance code in a manner similar to the standard gaseous fission products released from irradiated UO_2 fuel.

The mass of the released helium is given by

[ ] a, c (2)

where

[ ] a, c

and the total mass of helium released, M_{Helium}^{Total}, is obtained by a summation over the axial fuel rod nodes, N, which are coated with ZrB_2. The helium gas volume at STP is then computed from

$$V = M_{Helium}^{Total} * \bar{v} \tag{3}$$

where $\bar{v}$ is the specific volume from the Perfect Gas Law used in FATES3B

$$\bar{v} = \frac{RT}{P} = 6.205 * 10^5 \frac{inches^3}{lb-mole} \tag{4}$$

where

$$R = 1545 \frac{ft-lbs_f}{(lb-mole)^\circ R}$$
$$T = 492^\circ R$$
$$P = 14.7\,psia$$

Definition of the helium release fraction R_f [

.]^{a,c}

[

.]^{a,c}

2.3 ZRB$_2$ IFBA DESIGN AND LICENSING MODELS AND PROPERTIES

The required design and licensing models for ZrB_2 IFBA are simple and relatively straightforward. Implementation of ZrB_2 IFBA for helium release and the thermal and mechanical effects of the coating are described below. The CE implementation is similar to the implementation of the NRC-approved Westinghouse models.

2.3.1 Fuel Performance

The ZrB_2 depletion and the helium generation and release models described in Sections 2.1 and 2.2 are incorporated into the FATES3B fuel performance code. [

B_L^{10} R_f

.]^{a,c} As previously described, the released helium is added to the gap gas composition and the helium partial pressure is added to the fuel rod internal gas pressure.

In addition, the thickness of the ZrB_2 IBFA coating [

.]^{a,c}

[

.]^{a,c}

2.3.2 Safety Analysis Initial Conditions

The safety analyses (ECCS and non-LOCA) initial conditions, [

,]^{a,c} are based on the FATES3B data and predicted initial conditions prior to the assumed accident. Consequently, there are no changes required to the ECCS and non-LOCA codes and models due to the ZrB_2 IFBA.

2.3.3 Fuel Mechanical Design

Section 2.3.1 describes the incorporation of a new model in the FATES3B fuel performance code to account for the helium release associated with the burnout of the B-10. The resulting fuel rod internal pressures calculated by FATES3B are used as input to the mechanical design evaluations for stress, strain, fatigue, and collapse. Section 2.3.1 also describes the treatment of the ZrB_2 coating [

.][a,c] Since fuel rod internal pressure and initial fuel pellet diameter are handled the same as previously handled, no model changes are required in the mechanical design evaluations as a direct result of the ZrB_2 coating.

2.4 ANNULAR FUEL PELLET CONSIDERATIONS

The application of ZrB_2 IFBA may require the use of annular fuel pellets to provide additional void volume inside the fuel rod. Additional volume may be needed in order to meet maximum internal pressure limits, e.g., no-clad-lift-off. FATES3B incorporates annular fuel pellet capability as documented in the NRC approved fuel performance topical report, Reference 3. Although radial power and temperature distributions in annular fuel pellets provide thermal margin (i.e., lower temperatures) relative to solid fuel pellets at identical linear heat generation rates (LHGRs), the annular fuel pellets will be implemented only at the low power ends of the fuel rods (typically the top and bottom 5%, approximately). Therefore, the use of annular fuel pellets will not affect core operating margin. An evaluation of annular fuel pellets on ECCS evaluations and non-LOCA evaluations is discussed in Section 4. No annular fuel pellet models are required other than that in the FATES3B fuel performance code to determine internal hot gas pressures.

3.0 BENCHMARKING AND VERIFICATION

The benchmarking and verification of ZrB_2 IFBA is primarily through comparisons between computer code results to demonstrate that performance predictions will be similar within Westinghouse and CE designed PWRs.

3.1 NEUTRONICS

The presence of ZrB_2 as a thin coating on UO_2 fuel pellets in PWR fuel poses no additional requirements on the methods used for core neutronics design. Westinghouse currently has two neutronics design methodologies, each capable of accurately modeling the neutronics behavior of the ZrB_2 IFBA fuel. These are DIT-ROCS and PHOENIX-ANC, which are described in References 49, 53, 89, 90, 91, and 92. In addition, a third neutronics methodology, PARAGON-ANC (Reference 93), may be used to model core configurations containing ZrB_2 IFBA when PARAGON is approved by the NRC.

The neutron cross-sections of boron-10 are well known, and have been used in DIT and PHOENIX-P to compute the reactions of B-10 in soluble boron, in discrete burnable absorbers (Al_2O_3-B_4C and Wet Annular Burnable Absorbers, or WABAs), and in control rods. B-10 is relatively easy to calculate, unlike gadolinium and to a lesser degree erbium, and there are no unique requirements on spatial, spectral and depletion aspects of the calculation methods. The calculation of the neutronics of ZrB_2 IFBA is easier than that of the self-shielded burnable absorbers. A comparison of the references listed above shows that with respect to modeling features relevant to ZrB_2 IFBA, DIT is similar to PHOENIX-P, and that ROCS is similar to ANC.

PHOENIX-P and ANC are already licensed as the primary neutronic modeling tools for all Westinghouse reloads, most of which contain ZrB_2 IFBA. They have also been used for the reload analysis of CE designed PWRs (e.g., Fort Calhoun and Millstone 2), both with and without ZrB_2 IFBA. In addition, several benchmark comparisons between DIT-ROCS, PHOENIX-ANC on plants containing erbia, gadolinia, and ZrB_2 burnable absorbers has produced results that are essentially the same.

3.2 FUEL PERFORMANCE

The ZrB_2 IFBA depletion model is based on Westinghouse neutronics calculations as described in Section 2.1. Depletion and helium release incorporated in the FATES3B fuel performance code have been verified by a comparison to the Westinghouse PAD (Reference 95) results for the same fuel rod design and irradiation history. It can be seen, Figure 3-1, that the results are essentially identical.

Figure 3-1

PAD/FATES3B Comparison of IFBA Model

a, c

4.0 DESIGN AND LICENSING EFFECT OF ZRB_2 IFBA

4.1 EFFECT ON APPROVED TOPICAL REPORTS

The sections which follow provide a Roadmap discussion of the effect of ZrB_2 IFBA on CE design and safety analyses in the areas of fuel performance, fuel mechanical design, ECCS performance safety analysis for LOCA, non-LOCA transient analysis, and nuclear design. The implementation of the ZrB_2 IFBA is independent of cladding material and UO_2 models and properties, but NRC approval of the CE designed PWRs is currently, and will continue to be, limited to a peak pin average burnup of 60 MWd/kgU.

4.1.1 Fuel Performance

The current fuel performance models and methodology topical reports begin with Reference 38 as the base topical report. Additions and modifications to Reference 38 have been provided as supplements to augment the initial description. References 2 and 3 provided upgrades to the fuel performance code to reflect new performance data and extending models to higher burnups.

The currently approved fuel performance code FATES3B, References 2, 3, and 38, is supplemented by the ZrB_2 IFBA fuel helium generation and release models described in Section 2.0. This topical report, therefore, supplements References 2, 3, and 38.

The maximum internal pressure criterion report, Reference 11, previously supplemented the FATES3B topical reports. Reference 11 also provides fuel performance models for potential DNB propagation due to the higher internal gas pressures. However, no changes are required to the maximum pressure criterion, nor is there any direct impact of ZrB_2 IFBA on the fuel and cladding models in this approved topical, Reference 11. Reference 11 was supplemented with the ZIRLO™ cladding models of Reference 55. References 11 and 55 are unchanged because of the implementation of ZrB_2 IFBA or the need for annular pellets.

The gadolinia and erbia burnable absorbers are described in approved topical reports References 49 and 50 for gadolinia and Reference 53 for erbia. These topical reports also supplemented the FATES3B topical reports on the treatment of gadolinia and erbia in FATES3B. References 49, 50, and 53 are unchanged by the implementation of ZrB_2 IFBA fuel. The ZrB_2 IFBA treatment described herein supplements the FATES3B topical reports in a manner similar to the gadolinia and erbia burnable absorber topical reports as stated above.

In summary, the fuel performance topical reports are unchanged by the implementation of ZrB2 IFBA except as supplemented herein.

4.1.2 Fuel Mechanical Design

An assessment of the introduction and effect of ZrB_2 IFBA fuel on CE designed PWRs has determined that there is no effect on the fuel mechanical design. A review of applicable fuel mechanical design and licensing basis documents (References 12, 13, 42, 43, 47, 48, 54, and 55) was performed to determine the effect on fuel mechanical performance due to the implementation of the ZrB_2 IFBA fuel pellets. The survey has determined that there are no model changes required within fuel mechanical design in order to meet design criteria. [

.]a,c

4.1.3 ECCS Performance Evaluations

The versions of the Westinghouse ECCS Performance EMs for CE designed PWRs, with ZrB_2 IFBA fuel, are the 1999 Evaluation Model (1999 EM) for Large Break LOCA (LBLOCA) and the Supplement 2 Evaluation Model (S2M) for Small Break LOCA (SBLOCA). Table 4.1.3-1 lists the topical report references and the NRC's Safety Evaluation Reports (SERs) associated with the 1999 EM and the S2M.

The 1999 EM includes the following computer codes: CEFLASH-4A and COMPERC-II perform the blowdown and refill/reflood hydraulic analyses, respectively. In addition, COMPERC-II calculates the minimum containment pressure and FLECHT-based reflood heat transfer coefficients. STRIKIN-II performs the hot rod heatup analysis. COMZIRC, which is a derivative of the COMPERC-II code, calculates the core-wide cladding oxidation percentage. Refer to Table 4.1.3-1 for the references and SERs for these computer codes.

The S2M uses the following computer codes: CEFLASH-4AS performs the hydraulic analysis prior to the time that the Safety Injection Tanks (SITs) begin to inject. After injection from the SITs begins, COMPERC-II is used to perform the hydraulic analysis. COMPERC-II is used in the SBLOCA EM for larger break sizes which exhibit prolonged periods of SIT flow and significant core voiding. The hot rod heatup analysis is performed by STRIKIN-II during the initial period of forced convection heat transfer and by PARCH during the subsequent period of pool boiling heat transfer. Refer to Table 4.1.3-1 for the references and SERs for these computer codes.

The 1999 EM and S2M are NRC-accepted for ECCS performance analyses of CE designed PWRs fueled with either Zircaloy-4 or ZIRLO™ clad fuel assemblies.

A review of the documentation basis of the 1999 EM and the S2M listed in Table 4.1.3-1, which included a review of the respective SERs, identified and dispositioned the following potential issues with respect to applying the EMs to CE designed PWRs containing ZrB_2 IFBA fuel:

1. As required by the SER for the LBLOCA EM (Reference 72), the volumetric average fuel temperature at the maximum power location in the LOCA calculation (CEFLASH-4A and STRIKIN-II) must be equal to or greater than that calculated by the approved version of the FATES3B fuel performance code. Since the fuel pellet material properties in FATES3B do not require modification in order to analyze ZrB2 IFBA fuel, no changes to the ECCS EMs are required. The changes to FATES3B for the helium gas release and fuel rod internal pressure, and the addition of the ZrB2 coating thickness, are directly linked as input to the LBLOCA codes. Therefore, this SER constraint on the interface between the LBLOCA codes and the FATES3B fuel performance code continues to be met.
2. In the S2M, the hot rod heatup calculation is initialized at the burnup with the highest initial fuel stored energy. This approach may not yield a limiting peak cladding temperature for ZrB_2 IFBA fuel because of variations in the timing of cladding rupture due to the [ ]a,c in

the rod internal pressure of a ZrB_2 IFBA fuel rod at burnups near the burnup with the highest initial fuel stored energy. As described in Section 4.2.3.2, a parametric study of rod internal pressure is included in SBLOCA analyses to ensure that the potentially adverse influence of the timing of cladding rupture on peak cladding temperature (PCT) is captured in the analysis.

3. The fuel rod models in the 1999 EM and S2M computer codes assume the fuel pellet is solid and the fuel pellet stack is axially uniform. This precludes the ability to explicitly model annular fuel pellets in only the upper and lower extremities of the fuel pellet stack, if they are employed. The studies described in Section 4.2.3 demonstrate that explicit modeling of annular fuel pellets at the upper and lower extremities of the pellet stack [.][a,c]
4. The fuel pellet models in the EM computer codes [ ][a,c] for the effects of the ZrB_2 coating on the fuel pellet properties (e.g., specific heat, thermal conductivity, emissivity, etc.). [.][a,c]
5. The SER supporting the application of the 1999 EM and S2M to fuel designs with ZIRLO™ cladding (Reference 88) states that future changes to LOCA methodologies and/or constituent models require documentation supporting the change(s) that includes justification of the continued applicability of the methodology or model to ZIRLO™. There is no impact on the applicability of the methodology to analyze ZrB_2 IFBA fuel with ZIRLO™ cladding material.
6. The SER supporting the LBLOCA cladding rupture model in the 1999 EM (Reference 62) requires that the cladding rupture temperature be no higher than 950 °C (1742 °F) for fuel designs with Zircaloy-4 cladding. This SER constraint will continue to be met. This SER constraint does not apply to fuel rod designs with ZIRLO™ cladding.

4.1.4 Non-LOCA Transient Safety Analysis

The NRC-approved topical reports for non-LOCA transient safety analysis, References 28, 44, 45, 52, 57, 60, 71, and 75 were reviewed for this evaluation.

As discussed in Section 4.2.4 below, an evaluation was performed to determine if any of the changes associated with ZrB_2 IFBA would require a revision to current codes and methods used for the analysis of non-LOCA transient events. The review considered the effect of ZrB_2 IFBA implementation on core neutronics characteristics and on fuel mechanical design. It was determined that the current methodology remains valid for ZrB_2 IFBA fuel in CE designed PWRs.

4.1.5 Nuclear Design

The NRC-approved topical reports which address neutronics capability for the nuclear design of CE designed PWRs are Reference 89 for ROCS/DIT, References 90, 91, and 92 for PHOENIX and ANC. PARAGON, another neutronics methodology (Reference 93), is currently under NRC review. All have existing capability to treat the neutronic effects of ZrB_2 IFBA fuel. Application of gadolinia and erbia burnable absorbers in CE designed PWRs is provided by References 49, 50, and 53, which are also NRC-approved. Consequently, there are no neutronics models or methodology changes required to implement ZrB_2 IFBA fuel rod designs for CE designed PWRs.

Table 4.1.3-1

Topical Reports and Safety Evaluation Reports for the 1999 EM and the S2M

Subject	Topical Report Reference	SER Reference
LBLOCA Evaluation Model (CENPD-132)	14	72
Supplement 1	15	72
Supplement 2	16	74
Supplement 3	17	62
Supplement 4	18	82
SBLOCA Safety Evaluation Model (CENPD-137)	32	72
Supplement 1	33	70
Supplement 2	34	83
CEFLASH-4A (CENPD-133)	19	72
Supplement 2	21	72
Supplement 4	23	82
Supplement 5	24	62
CEFLASH-4AS		
Supplement 1 to CENPD-133	20	72
Supplement 3 to CENPD-133	22	70
COMPERC-II (CENPD-134)	25	72
Supplement 1	26	72
Supplement 2	27	62
STRIKIN-II (CENPD-135)	28	72
Supplement 2	29	72
Supplement 4	30	65
Supplement 5	31	80
PARCH (CENPD-138)	35	72
Supplement 1	36	72
Supplement 2	37	66
HCROSS		
Appendix A to Enclosure 1 to LD-81-095	56	62
COMZIRC		
Appendix C to CENPD-134 Supplement 1	26	72
Application of FLECHT Correlation to 16x16 Fuel Assemblies (CENPD-213)	46	67
Application of NUREG-0630 Cladding Rupture and Swelling Models (Enclosure 1 to LD-81-095)	56	62
Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs (CENPD-404-P-A)	55	88

4.2 ANALYSIS PROCEDURES AND METHODOLOGY

The sections which follow describe the typical effect of ZrB_2 IFBA fuel rod design on the design and safety analyses performance of CE designed PWRs.

4.2.1 Fuel Performance

4.2.1.1 Analysis

The analysis of ZrB_2 IFBA fuel and the comparisons to urania-erbia and UO_2 fuel presented in this section are intended to demonstrate the relative effect of the properties on various fuel performance parameters. Plant-specific evaluations were performed for reload analyses of cores which include the ZrB_2 IFBA fuel. The approach taken was to utilize typical CE fuel rod designs and to assume fuel rod power histories that typically bound anticipated operation. The power histories generally simulate operation to the core linear heat generation rate (LHGR) limits and, when applicable, to certain fuel rod design limits. For example, [

]^{a,c}

Analysis of ZrB_2 IFBA fuel, urania-erbia fuel, and UO_2 fuel in a standard reload analysis for a specific core may result in a predicted maximum internal hot gas pressure that is [ ]^{a,c} the design pressure limit.

4.2.1.2 Fuel Design

Current generation fuel rod designs typical of CE designed 14x14 and 16x16 fuel assembly fuel are evaluated and results presented. The characteristics of each fuel type analyzed are summarized in Table 4.2-1. ZrB_2 IFBA characteristics for the 14x14 and 16x16 fuel assembly designs summarized in Table 4.2-1 are representative of designs expected to be implemented [

.]^{a,c} The ZrB_2 IFBA fuel rod design for a specific reload application may differ from the demonstration designs of Table 4.2-1. Table 4.2-1 shows ZrB_2 IFBA fuel rod design parameters for eight representative designs. These designs include two ZrB_2 coated fuel rods for each of the 14x14 and 16x16 fuel assembly designs, i.e., one ZrB_2 IFBA fuel rod with all solid fuel pellets, and a second ZrB_2 IFBA fuel rod with annular fuel pellets in a short segment on each end of the fuel pellet stack (see the schematic in Figure 1-1). The designs also include a third urania-erbia fuel rod and a fourth UO_2 fuel rod each for the 14x14 and 16x16 fuel assembly designs.

4.2.1.3 Assumed Power Histories

Bounding power histories, based on the most limiting and highest expected B-10 loading design (the ZrB_2 IFBA with all solid pellets in these demonstration analyses were most limiting because of the high B-10 loading), were used in the evaluations for these typical 14x14 and 16x16 fuel assembly fuel rods. These power histories include use of [

]^{a,c}. Note, however, that cycle specific power histories are also used in the design and licensing if they bound the specific cycle. The bounding radial peaking factors for the 14x14 and 16x16 designs are shown in Figures 4.2-1 and 4.2.2.

The evaluations of the relative thermal performance of the 14x14 fuel rod designs consisted of comparing the ZrB_2 IFBA fuel rods with the urania-erbia and the UO_2 fuel rod thermal performances using identical input power histories. Similarly, the evaluations to compare relative thermal performance of the 16x16 fuel rod designs consisted of comparing the ZrB_2 IFBA fuel rods with the urania-erbia and the UO_2 fuel rod thermal performances using identical input power histories. The power history used for the 14x14 fuel rods is different than, but similar to, the power history used for the 16x16 fuel rods.

4.2.1.4 Results

<u>14x14 Design</u>

The fuel rod maximum internal hot gas pressures for the 14x14 ZrB_2 IFBA fuel rods, the urania-erbia fuel rod, and the UO_2 fuel rod [ ]^{a,c} are shown in Figure 4.2-3. [

.]^{a,c}

<u>16x16 Design</u>

The fuel rod maximum internal hot gas pressures for the 16x16 ZrB_2 IFBA fuel rods, the urania-erbia fuel rod, and the UO_2 fuel rod [ ]^{a,c} are shown in Figure 4.2-4. [

.]^{a,c}

4.2.1.5 B-10 Coating

The effect of the ZrB_2 coating is to increase the hot gas pressures due to the release of helium gas from the coating as the burnable absorber boron in the ZrB_2 coating is depleted. The representative ZrB_2 IFBA fuel rods evaluated herein had an enriched boron [

.]^{a,c}

4.2.1.6 Conclusions

It is concluded that the fuel performance of the ZrB_2 IFBA fuel rod design will satisfy the same performance criteria as required of the UO_2, erbia, and gadolinia fuel rod designs currently operating in CE designed PWRs.

4.2.2 Fuel Mechanical Design

Section 4.1.2 describes the influence of the ZrB_2 IFBA fuel pellets on the various aspects of the mechanical design of the fuel rods and fuel assemblies. As documented in that section, the mechanical design aspects that require evaluation are those that are a function of the fuel rod internal pressure or the initial fuel pellet diameter. The pertinent mechanical design topics are cladding stresses, cladding strain, cladding fatigue, and cladding collapse. Reference 55 (ZIRLO™ report) contains the most recent discussion of these topics (Sections 5.4.2, 5.4.3, 5.4.4, and 5.4.1, respectively). Evaluations of the effect of the ZrB_2 IFBA fuel pellets on each of these topics have been performed using typical 14x14 and 16x16 fuel rod design configurations. The evaluations are discussed below.

4.2.2.1 Cladding Stress

Cladding stress is affected by fuel rod internal pressure, but it is not affected by the fuel pellet diameter. Due to the NCLO maximum pressure criterion, the maximum fuel rod internal pressures are constrained to be comparable between the ZrB_2 IFBA fuel rods and the non-IFBA fuel rods. Since tensile cladding stresses are associated with maximum fuel rod internal pressures, the tensile cladding stresses of the ZrB_2 IFBA fuel rods and the non-IFBA fuel rods will be comparable. [

.]^{a,c} Evaluation of the effect of the [ ]^{a,c} minimum pressure on compressive cladding stresses demonstrated that both the 14x14 and 16x16 fuel rod designs continue to satisfy their cladding compressive stress criteria while accommodating the [ ]^{a,c} fuel rod internal pressures associated with the ZrB_2 IFBA fuel rods.

4.2.2.2 Cladding Strain

Cladding strain is a function of the fuel rod internal pressure, as well as the pellet-to-clad gap. With regard to the use of the ZrB_2 IFBA fuel pellets, only the effect of the increased fuel pellet diameter will be evaluated since high fuel rod internal pressures maximize cladding strain predictions and, as discussed above, the maximum rod internal pressures have not increased. The impact of the reduced pellet-to-clad gap has been evaluated for both the 14x14 and 16x16 fuel rod designs with ZrB_2 IFBA fuel pellets. The evaluations demonstrated that both fuel rod designs continue to satisfy their cladding strain criterion while accommodating the reduced pellet-to-clad gap associated with the ZrB_2 IFBA fuel pellets.

4.2.2.3 Cladding Fatigue

Cladding fatigue is also a function of both rod internal pressure and pellet-to-clad gap. Both [ ]^{a,c} rod internal pressures and reduced pellet-to-clad gaps increase predicted cladding cumulative fatigue damage factors. Therefore, the effects of both these parameters were included in the evaluation of the 14x14 and 16x16 fuel rod designs with ZrB_2 IFBA fuel pellets. The evaluations demonstrated that both fuel rod designs continue to satisfy their cladding fatigue criterion while accommodating the [ ]^{a,c} fuel rod internal pressures and the reduced pellet-to-clad gap associated with the ZrB_2 IFBA fuel designs.

4.2.2.4 Cladding Collapse

The reduced pellet-to-clad gap of the ZrB_2 IFBA pellets does not affect cladding collapse predictions, but [ ]^{a,c} initial rod internal pressures do. Evaluations of the cladding collapse times in the active fuel region of the rods were made with the [ ]^{a,c} rod internal pressures using the CEPAN computer code for both the 14x14 and 16x16 rod design. The evaluation demonstrated that the predicted collapse times for both designs were in excess of their required residence time. [.]^{a,c} Thus, cladding collapse is not a concern for the ZrB_2 IFBA fuel design.

4.2.2.5 Conclusion

The impact of the incorporation of ZrB_2 IFBA fuel pellets on the mechanical design aspects of the 14x14 and 16x16 fuel rods is presented above. The results of evaluations are included for cladding stresses, cladding strain, cladding fatigue, and cladding collapse. The evaluation of each topic has demonstrated that both the 14x14 and 16x16 fuel rod designs with ZrB_2 IFBA pellets satisfy the applicable design criteria.

4.2.3 ECCS Performance Evaluations

This section describes the application of the Westinghouse Emergency Core Cooling System (ECCS) Performance Evaluation Models (EMs) for CE designed PWRs to the analysis of ZrB_2 IFBA fuel for Large Break and Small Break Loss-of-Coolant Accidents (LBLOCA and SBLOCA).

Section 4.1.3 describes a survey of the ECCS performance analysis EMs that identifies the applicable licensing basis documents, limitations and constraints, and the fuel properties and behavior characteristics important to the implementation of ZrB_2 IFBA fuel for CE designed PWRs for both the LBLOCA and SBLOCA EMs.

Section 4.2.3.1 describes the approach for modeling ZrB_2 IFBA fuel for LBLOCA and Section 4.2.3.2 describes the approach for SBLOCA EMs. Conclusions regarding the implementation of ZrB_2 IFBA fuel in the CE LBLOCA and SBLOCA ECCS performance EMs are presented in Section 4.2.3.3.

The Westinghouse post-LOCA Long Term Cooling EM for CE designed PWRs (Reference 96) does not model a fuel rod to the level of detail that is affected by the implementation of ZrB_2 IFBA fuel. Consequently, the post-LOCA Long Term Cooling EM is unaffected and, therefore, not addressed herein.

As described in Sections 1.0 and 2.0 above, a ZrB_2 IFBA fuel rod contains UO_2 fuel pellets with a thin ZrB_2 coating on the fuel pellet surface. A ZrB_2 IFBA fuel rod consists of ZrB_2 coated fuel pellets over the majority of the fuel pellet stack with uncoated UO_2 fuel pellets at the top and bottom of the fuel pellet stack. Additionally, the UO_2 fuel pellets at the extreme ends of the fuel pellet stack may be of an annular design. The ECCS evaluations described below are based on this ZrB_2 fuel rod design concept.

4.2.3.1 Large Break Loss-of-Coolant Accident

ZrB_2 IFBA fuel is represented in LBLOCA ECCS performance analyses via normal code inputs. Also, the LBLOCA ECCS performance analysis process applies, as approved by the NRC, to ZrB_2 IFBA fuel. The following is a list of LBLOCA input parameters that represent the standard plant specific and design specific aspects pertinent to the introduction of ZrB_2 IFBA fuel:

- Fuel performance parameters such as pellet surface roughness, fission gas composition, initial centerline temperature versus linear heat rate, initial cladding and pellet dimensions, initial fuel rod internal pin pressure and gas volume distribution versus burnup are input through the link to the FATES3B fuel performance code and through other standard fuel specific computer code inputs.
- Similarly, physics parameters such as axial power shape, radial peaking and pin power census are input through standard physics related computer code inputs.

Demonstration analyses for typical ZrB_2 IFBA fuel rod designs for both 14x14 and 16x16 fuel assemblies show no significant change in PCT (typically < 50 °F change, which depends on the ZrB_2 IFBA fill gas pressure) and maximum cladding oxidation compared to non-ZrB_2 IFBA fuel rod designs. Implementation analyses are performed to determine the plant-specific impact of the ZrB_2 IFBA fuel.

The LBLOCA demonstration analyses were performed for both configurations of ZrB_2 IFBA fuel described above, that is, with and without annular fuel pellets at both ends of the fuel rod. The fuel performance characteristics of the designs with and without annular fuel pellets are represented by their FATES3B fuel performance data which are linked to the LBLOCA model. [

]^{a,c}:

1. [

]^{a,c}.

2. [

]^{a,c}.

4.2.3.2 Small Break Loss-of-Coolant Accident

Similar to LBLOCA analyses using the 1999 EM, ZrB_2 IFBA fuel is modeled via computer code inputs in SBLOCA analyses with the S2M. Consequently, no computer code changes are required to analyze ZrB_2 IFBA fuel..

As described in Section 4.2.1 above, because of the gas release associated with ZrB_2 IFBA fuel, the variation of fuel rod internal pressure with burnup is [ ]^{a,c} for a ZrB_2 IFBA fuel rod than it is for a non-ZrB_2 IFBA fuel rod (e.g., a UO_2 or erbia fuel rod), particularly at lower burnups. Also, to compensate for the [ ]^{a,c} gas release, the initial fill gas pressure for a ZrB_2 IFBA fuel rod is [ ]^{a,c} than that of a non-ZrB_2 IFBA fuel rod. For example, a typical fill gas pressure for a non-ZrB_2 IFBA CE fuel rod is approximately [ ]^{a,c} psia. In comparison, the fill gas pressure for a ZrB_2 IFBA CE fuel rod may be approximately [.]^{a,c}

For a SBLOCA analysis using the S2M, the hot rod heatup calculation is performed at the burnup for which the initial fuel rod stored energy is highest (Reference 32, page 18). Typically, this occurs at a burnup of approximately 500 to 1000 MWD/MTU. For the CE fuel rod design, the initial fuel rod internal pressure [ ]^{a,c} at such low burnups. For example, for a typical 14x14 fuel assembly, the initial fuel rod internal pressure for the hot rod changes by [ ]^{a,c} between 500 and 1000 MWD/MTU and by approximately [ ]^{a,c} between 0 and 8000 MWD/MTU. In contrast, the initial rod internal pressure increases by approximately [ ]^{a,c} between 500 and 1000 MWD/MTU for a ZrB_2 IFBA fuel rod with annular pellets. Likewise, it increases by [ ]^{a,c} between 0 and 8000 MWD/MTU. See Figures 4.2-3 and 4.2-4 for typical fuel performance characteristics.

Because of the [ ]^{a,c} in fuel rod internal pressure for ZrB_2 IFBA fuel at low burnup and [ ]^{a,c}, the hot rod heatup calculation of a ZrB_2 IFBA fuel rod may show [ ]^{a,c} differences in PCT over a [ ]^{a,c} range of burnups.

As a result, a hot rod heatup calculation performed at the burnup with the maximum initial fuel stored energy may not be limiting. For example, a hot rod heatup calculation performed at an earlier burnup with [ ]^{a,c} may result in cladding rupture being delayed until later in the hot rod heatup transient when the cladding temperature is approaching its peak value. If the cladding temperature at this delayed rupture time is above the threshold temperature for cladding oxidation, the rupture may produce a rapid increase in cladding temperature due to the oxidation process.

A parametric study of rod internal pressure is included in SBLOCA analyses to ensure that the potentially adverse impact of the timing of cladding rupture on peak cladding temperature described above is captured in SBLOCA analyses. The limiting break is first identified by means of the break spectrum analysis, which is performed at the burnup corresponding to the maximum initial fuel rod stored energy. The parametric study is then performed to determine if a rod internal pressure different from the pressure at the burnup with the maximum initial fuel stored energy results in an increase in peak cladding temperature for the limiting break. In particular, the pool-boiling hot rod heatup calculation for the limiting break of the break spectrum is reanalyzed over the range of rod internal pressures identified by the hot rod fuel performance analysis. A sufficient number of rod internal pressures is analyzed in the parametric study to ensure that, if cladding rupture is predicted to occur for the limiting break, it occurs at a time that results in the maximum peak cladding temperature. To the extent required for a plant-specific analysis, the parametric study is performed for each fuel design covered by the analysis (e.g., ZrB_2 IFBA fuel rod and UO_2 fuel rod; Zircaloy-4 cladding and ZIRLO™ cladding).

A SBLOCA analysis of a typical ZrB_2 IFBA fuel rod design shows that, excluding the potential impact of the fuel rod internal pressure parametric study, implementation of ZrB_2 IFBA has an insignificant effect (i.e., < 50 °F change) on PCT, whereas including the impact of the parametric study may have a significant effect (i.e. > 50 °F). Implementation analyses are performed to determine the plant-specific impact of ZrB_2 IFBA fuel.

<u>Annular Fuel Pellets</u>

[

.]^{a,c}

4.2.3.3 Conclusions

EM surveys for both LBLOCA and SBLOCA have been conducted and the influence of the introduction of ZrB_2 IFBA fuel on the methodology basis has been addressed. Westinghouse concludes that no changes to the 1999 EM or S2M computer codes are required to implement ZrB_2 IFBA fuel, including ZrB_2 IFBA fuel rod designs that contain annular fuel pellets.

For LBLOCA, the gap conductance and internal fuel pin pressure models receive relevant interface data or initial conditions for ZrB_2 IFBA fuel through the link to FATES3B fuel performance code in the same manner as for non-ZrB_2 IFBA fuel. For SBLOCA, these aspects of the fuel pellet model are controlled through computer code inputs in the same manner as for non-ZrB_2 IFBA fuel.

For a ZrB_2 coated fuel pellet, material properties such as thermal conductivity, emissivity, and density are modeled [,][a,c] as described in Sections 2.0 and 2.3.

Evaluation model surveys for both LBLOCA and SBLOCA demonstrate that current SER constraints and limitations continue to apply, as described in Section 4.1.3.

Special studies were conducted for both LBLOCA and SBLOCA that show that annular pellet regions at the top and bottom of the ZrB_2 IFBA fuel rod can be represented [

.][a,c]

4.2.4 Non-LOCA Transient Safety Analysis

This section addresses the effect of the implementation of ZrB_2 IFBA fuel on the non-LOCA accident analyses. ZrB_2 IFBA related changes were evaluated to determine if any of the changes would require a revision to current codes and methods used for the analysis of non-LOCA events. It was determined that the current methodology remains valid for IFBA cores.

The evaluation included consideration of the following IFBA-related effects:

4.2.4.1 Changes to Core Neutronics Characteristics

Core Peaking

Core axial and radial peaks are an input to the non-LOCA safety analyses. An important effect of ZrB_2 IFBA implementation on the non-LOCA transient safety analyses is through the effect on core power peaking. Section 3.1 discusses the impact of ZrB_2 IFBA implementation on power peaking. The effect is relatively small and any change in core power peaking due to implementation of ZrB_2 IFBA will be accommodated in the same way as normal cycle-to-cycle changes.

Burnup Dependence of MTC

As a result of the more rapid burnout characteristics of ZrB2 IFBA, peak soluble boron concentration may occur sometime after beginning-of-cycle (BOC). As a consequence, peak positive MTC may occur later than BOC. However, non-LOCA transient safety analyses use bounding values of MTC that bound all times in core life. The bounding values remain valid for the ZrB2 IFBA fuel design.

4.2.4.2 Fuel Mechanical Design Characteristics

Decrease in Fuel Gas Gap

The ZrB2 IFBA fuel pellets will have a slightly larger radius than the standard UO2 fuel pellets so that the gas gap at BOC will be smaller. This will have a small effect on the gap heat conductance. Non-LOCA safety analyses use values of the gap conductance that bound all times in core life. The bounding values remain valid for the ZrB2 IFBA fuel design.

Gas Release

As discussed in Section 2.2, helium gas release occurs for the ZrB_2 IFBA fuel design. However, this is not a significant parameter for the non-LOCA transient safety analyses, and does not impact the results of the non-LOCA transient analyses.

Annular Fuel Pellets

The ZrB_2 IFBA fuel rod design may include a region of annular pellets at the top and bottom of the fuel rod. This feature is discussed in Section 2.4 above. The purpose of the annular fuel pellet region is to provide void volume to accommodate gas released by the burnup of B-10.

A review was performed to determine if the annular pellet region could be limiting for any of the design basis non-LOCA transient events. The review determined that the annular region would never be limiting. Consequently, the current methodology, which models the solid pellets, remains valid. This conclusion was based on the following considerations:

- The bounding core properties used as input to the non-LOCA transient analyses remain valid for the ZrB2 IFBA fuel design including annular fuel pellets.
- Thermal hydraulic behavior of the annular fuel region is unchanged. Therefore, the results of events that use DNBR as a criterion are not affected.
- It is expected that the annular fuel pellet design will be less likely than the solid fuel pellet design to induce cladding failure during energy insertion transients.
- The power in the annular fuel region will be well below that of the peak power in the solid fuel for all conditions of normal operation and transients.
- It was determined that only the CEA Ejection event could be potentially be impacted by the annular fuel pellets. However, an evaluation of the CEA ejection accident found that the deposited energy and temperature in the annular pellet region was significantly lower than the values obtained for the solid pellet region due to the lower power peaking in the annular fuel.

4.2.4.3 Conclusions

A review of the non-LOCA licensing basis analyses for CE designed PWRs was performed. It was determined that the current methodology remains valid for the analysis of ZrB_2 IFBA fuel and, furthermore, will provide bounding results for the ZrB_2 IFBA design.

The effect of the ZrB_2 IFBA design on the results of the non-LOCA transient analyses is small and will be accommodated in the same way as normal cycle-to-cycle changes.

4.2.5 Nuclear Design

This section describes the impact of ZrB_2 IFBA on the nuclear aspects of core design.

In general the behavior of a core with ZrB_2 IFBA is similar to that of a core with erbium burnable absorber, except that ZrB_2 IFBA exhibits no special spectral interaction with moderator temperature. Thus, a greater BOC reactivity hold-down and associated lower soluble boron is required with ZrB_2 IFBA to achieve the same MTC as with erbium. Since ZrB_2 IFBA burns out completely, additional ZrB_2 IFBA can be added as necessary to control MTC, without an ore/SWU penalty.

While ZrB_2 IFBA burns out a little faster than does erbium, ZrB_2 IFBA does not exhibit the extremely rapid burnout that is sometimes observed with low concentrations of gadolinium. Power peaking factors are similar between ZrB_2 IFBA and erbium, and usually lower than what can be achieved with gadolinium, for the same number of feed assemblies. The primary macroscopic characteristics of a core using ZrB_2 IFBA are a lower required soluble boron concentration at BOC and a lower average feed enrichment.

While the ZrB_2 IFBA fuel rods could be composed of all solid UO_2 pellets, it is common for ZrB_2 IFBA fuel rods to incorporate a small region of annular pellets at each end of the fuel stack. This design feature helps reduce the peak internal pressure, as described earlier in this report. The axial power for such regions is less than the average axial power, even without the use of axial blankets (typically at lower U-235 enrichment and lower power). With axial blankets the power in the annular region would be, therefore, substantially less than the average axial power.

In addition to the natural tendency for power to be lower near the ends of the core, the reduced mass of UO_2 in an annular fuel pellet results in an additional power offset relative to a nearby solid pellet. That is, for approximately the same incident fine-group neutron spectrum, the annular fuel produces less power. This power reduction is of the order of the volumetric fuel displacement.

Table 4.2-1
Fuel Rod Design Parameters

a, c

Table 4.2-1 (continued)
Fuel Rod Design Parameters

a, c

Figure 4.2-1 Maximum Allowable Radial Peaking Factor 14x14 Fuel Design

a, c

Figure 4.2-2 Maximum Allowable Radial Peaking Factor 16x16 Fuel Design

a, c

Figure 4.2-3 Maximum Internal Gas Pressure 14x14 Fuel Design

a, c

Figure 4.2-4 Maximum Internal Gas Pressure 16x16 Fuel Design

a, c

5.0 CONCLUSIONS

The ZrB_2 IFBA fuel rod design consists of a ZrB_2 coating on the outer diameter of UO_2 fuel pellets over the center region of the fuel rod with cutback regions (regions without ZrB_2 coating) on both ends of the fuel rod. Lower enrichment fuel pellets may also be used in a portion of the cutback region. The cutback regions may consist of solid, annular, or a solid and annular fuel pellet combination as described in Section 1.1.

ZrB_2 helium gas generation and release are incorporated into the FATES3B fuel performance code in a manner similar to the approved Westinghouse PAD implementation of ZrB_2 IFBA. Thickness of the ZrB_2 coating is accounted for in the fuel-clad gap and mechanical interaction models where appropriate. Neutronic codes already contain the capability to model ZrB_2 IBFA fuel rods. An engineering evaluation was performed for the impact of ZrB_2 IFBA on fuel rod design and safety analyses in the areas of fuel performance, fuel mechanical design, ECCS performance evaluations, non-LOCA transient safety analyses, and neutronic design. No significant issues were found to exist.

Consequently, the ZrB_2 IFBA fuel rod design can be implemented for CE designed PWRs on a full batch basis without significant design and licensing perturbations.

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Appendix A

Nuclear Regulatory Commission

Round #1 Request for Additional Information

dated July 10, 2003

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NRC Round #1 Request for Additional Information on WCAP-16072-P, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"

1. Section 5 best describes the ZrB_2 IFBA fuel rod design as consisting of "...a ZrB_2 coating on the outer diameter of UO_2 fuel pellets over the center region of the fuel rod with cutback regions (regions without ZrB_2 coating) on both ends of the fuel rod." The description continues, "Lower enrichment fuel pellets may also be used in a portion of the cutback region... The cutback regions may consist of solid, annular, or a solid and annular fuel pellet combination..."
   a. Evaluations credit the location of annular fuel pellets in the lower power ends of the fuel stack. Yet, the topical does not provide any limitation on the axial length of the annular pellet cutback regions. Please provide the supporting technical basis for your conclusion.
   b. The topical states that the ZrB_2 IFBA coating may be natural or enriched with the B^{10} isotope. Yet, the topical does not provide any limitation on the extent of B^{10} enrichment or its potential impact on core physics predictions. Please provide the supporting technical basis for your conclusion.
   c. Evaluations credit the narrow width of the ZrB_2 coating. The topical states that the coating thickness can vary within a specified range. Please describe the impact of coating thickness on core physics predictions.
   d. The IFBA fuel rod design includes lower U^{235} enrichment axial blanket regions. The topical does not provide any detail on limitations on the axial length or the enrichment split of these blanket regions. Please provide additional information on these axial blanket regions.

2. The topical does not provide any information on the impact of U^{235} enrichment axial blanket regions on core physics predictions and safety analyses. Please provide the analyses that demonstrate that the impact of these axial blanket regions are acceptable.

3. The topical does not provide any information on the impact of biases/uncertainties and manufacturing tolerances on the ZrB_2 coating thickness and B^{10} enrichment on core physics predictions and safety analyses. Please provide the analyses that demonstrate that these uncertainties and tolerances have been properly accounted for.

4. Section 1.2 states, "...Westinghouse has had considerable fabrication and operational experience with the ZrB_2 Integral Fuel Burnable Absorber." Section 1.3 states, "Post-irradiation examinations of ZrB_2 IFBA test rods revealed no profilometry anomalies in the coated fuel pellet zone, no chemical interaction between the coating and fuel rod cladding, no incipient cracks in the cladding inner diameter, no excessive fuel pellet cracking, nor any anomalies in the fuel structure."
   a. Please provide details of Westinghouse's fabrication and operational experience with annular fuel pellets.
   b. Please provide details of Westinghouse's fabrication and operational experience with U^{235} enrichment axial blanket regions.
   c. Please provide details of post-irradiation examinations of annular fuel pellets.

5. With regard to the modeling capability of the CE design analyses:

   a. Is FATES3B capable of specifically modeling the different axial regions of the IFBA fuel rod (e.g., annular vs. solid pellet region, enrichment blankets, ZrB_2 coating and cutback regions)?

   b. Will the axial nodes be aligned in such a way as to avoid splitting these different axial regions?

   c. What are the limitations of STRIKIN-II with respect to calculating radial power distribution in an annular pellet?

   d. In the annulus region, how does FATES3B model relocation, thermal expansion, and swelling? Is there an experience database available to validate these models specifically for annular fuel?

6. In order to compensate for the helium production associated with the B-10 depletion, the initial helium fill pressure will be reduced. One consequence of this change would be a lower BOC gap conductivity. Does this reduced gap conductivity remain above the minimum gap conductivity assumed in the UFSAR safety analyses for each of the CE plants?

7. Section 2.2 states, "the maximum and minimum ZrB_2 IFBA helium release will be applied deterministically consistent with the specific applications." Please provide the values and bases for the minimum and maximum helium release fraction.

8. Section 4.2.2.4 states, "The evaluations of cladding collapse in the plenum region of the rods demonstrated that cladding collapse would not occur if the radial support offered to the cladding by the plenum spring is factored into the analysis." Is credit for the spring required to compensate for potential negative attributes of the IFBA fuel rod design?

9. Section 4.2.3 states, "...the post-LOCA Long Term Cooling EM is unaffected and, therefore, not addressed herein." IFBA has the potential to influence the initial critical boron concentration of the RCS which in turn may impact the timing and magnitude of boron precipitation in the LTC Analysis. Please provide the supporting technical basis for your conclusion.

10. Section 4.2.4.1 discusses the impact of ZrB_2 IFBA on core peaking. Does the rapid depletion of B-10 result in an increasing radial peaking factor in the beginning of cycle (e.g. Fr increases with burn up initially, then burns down)?

11. Section 4.2.4.1 states, "peak soluble boron concentration may occur sometime after beginning-of-cycle:"

   a. Will Plant Operations staff be trained and will Operating Procedures be updated to reflect this new operating scheme?

   b. Plant Technical Specifications require MTC surveillance tests to validate the physics predictions (and safety analyses) and ensure that plant operations remain within Technical Specification limits. Please justify how TS SR 3.1.3.1 and SR 3.1.3.2 perform their intended purpose in the presence of an increasing MTC at startup.

12. Figures 4.2-3 and 4.2-4 illustrate higher rod internal pressures with ZrB_2 IFBA. These higher internal rod pressures, especially at lower burnup, translate into a greater challenge to DNB Propagation. With the current methodology, a greater number of pins is likely to balloon and even rupture during Non-LOCA events which experience DNB. Please provide the analyses that demonstrate that this further challenge to DNB Propagation is acceptable from a radiological dose perspective, core coolable geometry perspective, and fuel relocation perspective.

13. The annulus region of an IFBA fuel rod (with annular fuel pellets) may potentially be filled with moderator as the result of a clad failure. Have any evaluations been completed to assess the neutronic, thermal, and mechanical behavior of a flawed IFBA fuel rod under normal, transient, and shutdown (including spent fuel pool) conditions?

14. Section 4.2.3.2 states, "the pool-boiling hot rod heatup calculation for the limiting break of the break spectrum is reanalyzed over the range of rod internal pressures identified by the hot rod fuel performance analysis. A sufficient number of rod internal pressures is analyzed..." Please quantify the range of pressures evaluated and the technical basis of this range.

15. As a follow on to RAIs 4c and 5d, please provide analyses that demonstrate that the fuel relocation models are capable of accurately accounting for cladding ruptures within or just below the annular fuel region.

Request for Additional Information Concerning Dose Calculations

The Westinghouse safety analysis provided in WCAP-16072-P, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," addresses the impact on fuel rod design and safety analyses in the areas of fuel performance, fuel mechanical design, ECCS performance evaluations, non-LOCA transient safety analyses, and neutronic design. The safety analysis does not address the impact of the proposed changes (i.e., increased fuel rod pressure, annular fuel pellets, and/or axial blankest) on the generation of fission products and the transport of fission products released during LOCA and non-LOCA design basis accidents. These considerations need to be addressed in order for the staff to make a finding that the use of the proposed fuel will not impact assumptions used in the current licensing basis analysis for demonstrating compliance with 10 CFR Part 100, or 10 CFR 50.67, as applicable. For example:

1. Does the addition of the zirconium diboride coating alter the source term assumptions provided in NUREG-1465 with regard to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release? In particular, does the added zirconium diborate upset current conclusions regarding the radiochemistry of iodines (e.g., does the increased mass of Zr shift the chemical equilibrium of reactions that yield CsI as opposed to ZrI?)

2. Does the increased helium gas pressure in the fuel pins invalidate assumptions regarding fuel handling accident fission product release iodine scavenging in the spent fuel pool or reactor cavity? Note that the pool decontamination factors in Safety Guide 25, "Assumptions Used for Evaluating

the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," are predicated on a maximum fuel rod pressurization of 1200 psig. (The assumptions in this safety guide were largely developed from the results of experiments performed by Westinghouse as reported in WCAP-7518-L.)

3. Is the fission product migration in an annular pellet different from that of a solid pellet in a manner that would affect previous analysis assumptions related to the fraction of core inventory in the fuel gap, or on the timing of fission product releases following a transient? Is the correlation between burnup and fission product diffusion different for an annular pellet than a solid pellet?

These three questions are offered only as examples. The staff expects Westinghouse to provide a full evaluation of the impact of zirconium diboride on the generation and transport of fission products as currently analyzed in DBA radiological consequence analyses.

Appendix B

Westinghouse Electric Company LLC Response to

Nuclear Regulatory Commission

Round #1 Request for Additional Information

LTR-NRC-03-57, dated September 10, 2003

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Project No.: 700
Our ref: LTR-NRC-03-57

September 10, 2003

Subject: **"Response to NRC Request for Additional Information WCAP-16072-P & -NP, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"" (Proprietary / Non-proprietary)**

References:
1. Fax, B. J. Benney (NRC) to R. Sisk (W), "WCAP-16072 Formal RAIs", July 10, 2003
2. WCAP-16072-P & -NP, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs", April 2003
3. Letter, H. A. Sepp (W) to USNRC Document Control Desk, "Submittal of WCAP-16072-P, Revision 0, Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs, (Proprietary/Non-proprietary)", LTR-NRC-03-14, April 25, 2003

Enclosed are copies of Westinghouse Electric Company LLC (Westinghouse) responses to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI), Reference 1, regarding WCAP-16072-P & -NP, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs", Reference 2. This topical report was submitted for NRC review and approval on April 25, 2003, Reference 3.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-03-1701 with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit, AW-03-1701.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-03-1701 and should be addressed to H. A. Sepp, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz, NRC (w/o enclosures)
B. J. Benney, NRC (w/ 3 proprietary & 1 non-proprietary copies)
P. Clifford, NRC (w/o enclosures)
D. G. Holland, NRC (w enclosures)
E. S. Peyton, NRC (w enclosures)

Enclosure

Response to NRC Request for Additional Information

WCAP-16072-P & -NP

"Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"

RAI No. 1a:

Section 5 best describes the ZrB_2 IFBA fuel rod design as consisting of "...a ZrB_2 coating on the outer diameter of UO_2 fuel pellets over the center region of the fuel rod with cutback regions (regions without ZrB_2 coating) on both ends of the fuel rod". The description continues, "Lower enrichment fuel pellets may also be used in a portion of the cutback region....The cutback regions may consist of solid, annular, or a solid and annular fuel pellet combination...".

Evaluations credit the location of annular fuel pellets in the lower power ends of the fuel stack. Yet, the topical does not provide any limitation on the axial length of the annular pellet cutback regions. Please provide the supporting technical basis for your conclusion.

Response 1a:

[

]^{a,c} Although the exact length may vary depending on what is required to provide optimal peaking for individual plants or cycles, it is anticipated that [

]^{a,c} No specific limitation on the size of this region is necessary since core design guidelines and cycle specific calculations will explicitly verify that the required power margin in the annular pellet region is maintained.

RAI No. 1b:

The topical states that the ZrB_2 IFBA coating may be natural or enriched with the B^{10} isotope. Yet, the topical does not provide any limitation on the extend of B^{10} enrichment or its potential impact on core physics predictions. Please provide the supporting technical basis for your conclusion.

Response 1b:

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]^{a,c}

RAI No. 1c:

Evaluations credit the narrow width of the ZrB_2 coating. The topical states that the coating thickness may vary within a specified range. Please describe the impact of coating thickness on core physics predictions.

Response 1c:

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]^{a,c}

RAI No. 1d:

The IFBA fuel rod design includes lower U^{235} enrichment axial blanket regions. The topical does not provide any detail on limitations on the axial length or the enrichment split of these blanket regions. Please provide additional information on these axial blanket regions.

Response 1d:

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]^{a,c}

As stated above, the cycle specific reload calculations will verify that the required power margin in the annular pellet region is maintained even in cases where the blankets are fully enriched.

RAI No. 2:

The topical does not provide any information on the impact of U^{235} enrichment axial blanket regions on core physics predictions and safety analyses. Please provide the analyses that demonstrate that the impact of these axial blanket regions are acceptable.

Response 2:

Low enriched blankets have been used extensively in PWR plants throughout the US. Most of the Westinghouse plants currently employ low enriched axial blankets. The Westinghouse physics code ANC has been extensively benchmarked to cycles containing low enriched axial blankets. The PHEONIX/ANC code package is currently being used to perform reload analysis for St. Lucie 2 core, a CE plant that contains low enriched axial blankets. Axial blankets have been used in the St. Lucie 2 core for three (3) cycles. The St. Lucie 2 UFSAR was appropriately updated. Other CE plants have indicated an interest in axial blankets as well.

The DIT/ROCS computer code systems has been used to analyze several of these cycles. Topical report CENPD-275-P-A contains results of benchmarks of the DIT/ROCS computer codes on St. Lucie Unit 1 Cycle 7 and St. Lucie Unit 2 Cycle 3 that contained several LTAs with low enriched axial blankets. In all of these cases, no significant impact on the accuracy of the physics predictions was observed.

The impact of low enriched axial blankets will be explicitly considered in the plant specific safety analysis. The physics analysis that will be performed to support the implementation of the axial blankets will explicitly model the low enriched axial blankets. The impact of these blanket regions on parameters important to safety will be calculated and used to revise the safety analysis where necessary.

RAI No. 3:

The topical does not provide any information on the impact of biases/uncertainties and manufacturing tolerances on the ZrB_2 coating thickness and B^{10} enrichment on core physics predictions and safety analyses. Please provide the analyses that demonstrate that these uncertainties and tolerances have been properly accounted for.

Response 3:

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]a,c

The impact of uncertainty in B^{10} loading on helium release and internal pressure is conservatively accounted for in a manner as discussed in Response 7.

RAI No. 4:

Section 1.2 states, "...Westinghouse has had considerable fabrication and operational experience with the ZrB_2 Integral Fuel Burnable Absorber". Section 1.3 states, "Post-irradiation examinations of ZrB_2 IFBA test rods revealed no profilometry anomalies in the coated fuel pellet zone, no chemical interaction between the coating and fuel rod cladding, no incipient cracks in the cladding inner diameter, no excessive fuel pellet cracking, nor any anomalies in the fuel structure".

4 Please provide details of Westinghouse's fabrication and operational experience with annular fuel pellets.

5 Please provide details of Westinghouse's fabrication and operational experience with U^{235} enrichment axial blanket regions.

6 Please provide details of post-irradiation examinations of annular fuel pellets.

Response 4:

Summation of IFBA Rods, Axial Blanket (AB) Pellet Rods and Annular Axial Blanket (AAB) Pellet Rods per Year

Year	# of IFBA Rods	# of Rods with AB		# of Rods with AAB	
		Natural	Mid-enriched	Natural	Mid-enriched or Fully-enriched

a, b

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]a,c

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]^{a,c}

RAI No. 5a:

With regard to the modeling capability of the CE design analyses:

Is FATES3B capable of specifically modeling the different axial regions of the IFBA fuel rod (e.g. annular vs solid pellet region, enrichment blankets, ZrB_2 coating and cutback regions)?

Response 5a:

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]a,c

RAI No. 5b:

Will the axial nodes be aligned in such a way as to avoid splitting these different axial regions?

Response 5b:

The actual lengths of annular pellets and ZrB_2 coatings are expected to coincide closely with FATES3B axial node lengths. However, if they do not, the input and FATES3B adjustments are described in Response 5a.

RAI No. 5c:

What are the limitations of STRIKIN-II with respect to calculating radial power distribution in an annular pellet?

Response 5c:

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]^{a,c} Solid and annular pellet radial power distributions are described in approved FATES3B reports, Reference 38 and References 2 and 3, respectively, of WCAP-16072-P.

RAI No. 5d:

In the annulus region, how does FATES3B model relocation, thermal expansion, and swelling? Is there an experience database available to validate these models specifically for annular fuel?

Response 5d:

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RAI No. 6:

In order to compensate for the helium production associated with the B^{10} depletion, the initial helium fill pressure will be reduced. One consequence of this change would be a lower BOC gap conductivity. Does this reduced gap conductivity remain above the minimum gap conductivity assumed in the UFSAR safety analyses for each of the CE plants?

Response 6:

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]^{a,c}

RAI No. 7:

Section 2.2 states, "...the maximum and minimum ZrB_2 IFBA helium release will be applied deterministically consistent with the specific applications". Please provide the values and bases for the minimum and maximum helium release fraction.

Response 7:

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]^{a,c}

RAI No. 8:

Section 4.2.2.4 states, "The evaluations of cladding collapse in the plenum region of the rods demonstrated that cladding collapse would not occur if the radial support offered to the cladding by the plenum spring is factored into the analysis". Is credit for the spring required to compensate for attributes of the IFBA fuel rod design?

Response 8:

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]a,c

RAI No. 9:

Section 4.2.3 states, "...the post-LOCA Long Term Cooling EM is unaffected and, therefore, not addressed herein". IFBA has the potential to influence the initial critical boron concentration of the RCS which in turn may impact the timing and magnitude of boron precipitation in the LTC Analysis. Please provide the supporting technical basis for your conclusion.

Response 9:

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]a,c

The maximum RCS boron concentration is determined by the cycle length and the burnable absorber worth loaded into the core. The required burnable absorber worth for CE plants is typically set by number of burnable absorber rods required to reduce the RCS boron concentration to a value low enough to assure that the most positive MTC Tech Spec limit will not be exceeded. [

]a,c

RAI No. 10:

Section 4.2.4.1 discusses the impact of ZrB_2 IFBA on core peaking. Does the rapid depletion of B-10 result in an increasing radial peaking factors in the beginning of cycle (e.g. Fr increases with burnup initially, then burns down)?

Response 10:

Cores containing ZrB_2 IFBA often experience the highest radial peak sometime after BOC. This is also true for most cores containing gadolinia burnable absorber and for some cores containing the erbia burnable absorber. This behavior is thus not unusual and is not anticipated to cause any problems. The current CE safety analysis methodology does not assume or require that the power peaking be monotonically decreasing.

RAI No. 11a:

Section 4.2.4.1 states, "peak soluble boron concentration may occur sometime after beginning-of-cycle".

Will Plant Operations staff be trained and will Operating Procedures be updated to reflect this new operating scheme?

Response 11a:

This is a plant specific implementation issue, but will be recommended to the utility. Some cycle designs may show a slow and modest increase in the RCS critical boron concentration over the first third of the cycle. When ZrB_2 IFBA is implemented in a specific plant, the utility may decide that some training is necessary in order to alert the plant operations staff to the possibility of this behavior for some cycles. The utility may decide to review and update plant operating procedures to accommodate potential differences in core behavior between ZrB_2 and erbia IFBAs.

RAI No. 11b:

Plant Technical Specifications require MTC surveillance tests to validate the physics predictions (and safety analyses) and ensure that plant operations remain within Technical Specification limits. Please justify how TS SR 3.1.3.1 and SR 3.1.3.2 perform their intended purpose in the presence of an increasing MTC at startup.

Response 11b:

This is a plant specific implementation issue. The Tech Specs currently require that the MTC be measured at HZP, at HFP near BOC (for some plants this has been described as prior to reaching 800 ppm RCS boron), and prior to reaching two thirds of the cycle length. Note that currently only the HZP BOC MTC measurement is used to confirm that the MTC is less than the most positive limit. Thus no explicit changes to the Tech Specs are necessary. However Westinghouse will recommend that procedures be implemented to confirm that the MTC (either by direct measurement or by extrapolation from other cycle specific measurements) is within its limits at the highest RCS boron concentration expected during the cycle.

RAI No. 12:

Figures 4.2-3 and 4.2-4 illustrate higher rod internal pressures with ZrB_2 IFBA. These higher internal rod pressures, especially at lower burnup, translate into a greater challenge to DNB Propagation. With the current methodology, a greater number of pins is likely to balloon and even rupture during Non-LOCA events which experience DNB. Please provide the analyses that demonstrate that this further challenge to DNB Propagation is acceptable from a radiological dose perspective, core coolable geometry perspective, and fuel relocation perspective.

Response 12:

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]a,c

RAI No. 13:

The annulus region of an IFBA fuel rod (with annular fuel pellets) may potentially be filled with moderator as the result of a clad failure. Has any evaluations been completed to assess the neutronic, thermal, and mechanical behavior of a flawed IFBA fuel rod under normal, transient, and shutdown (including spent fuel pool) conditions?

Response 13:

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]a,c

The current methodology used for spent fuel pool criticality analysis conservatively assumes that the annular pellet regions are comprised of solid, full diameter pellets. This is conservative since it results in the highest K_{eff} for the fuel storage configuration. The reactivity calculated for this configuration would bound the reactivity associated with a fuel rod comprised of annular pellets containing water in the annulus.

Westinghouse's operating experience with behavior of ZrB_2 IBFA fuel rods does not indicate any difference in behavior from non-IFBA fuel rods. Furthermore, operation with potentially flawed or failed fuel rods is a very low probability. No requirement exists to specifically model such rods. Since the probability of the existence of such flawed/failed fuel is small, e.g., 1 or 2 rods at most in the core, the impact on normal, transient, or shutdown conditions would be insignificant. Consequently, no specific evaluations have been performed and Westinghouse believes that such calculations are not needed.

RAI No. 14:

Section 4.2.3.2 states, "the pool-boiling hot rod heatup calculation for the limiting break of the break spectrum is reanalyzed over the range of rod internal pressures identified by the hot rod fuel performance analysis. A sufficient number of rod internal pressures is analyzed...". Please quantify the range of pressures evaluated and the technical basis of this range.

Response 14:

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]a,c

RAI No. 15:

As a follow on to RAIs 4c and 5d, please provide analyses that demonstrate that the fuel relocation models are capable of accurately accounting for cladding ruptures within or just below the annular fuel region.

Response 15:

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Additional Information Concerning Dose Calculations

The Westinghouse safety analysis provided in WCAP-16072-P, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," addresses the impact on fuel rod design and safety analyses in the areas of fuel performance, fuel mechanical design, ECCS performance evaluations, non-LOCH (sic) transient safety analyses, and neutronic design. The safety analysis does not address the impact of the proposed changes (i.e., increased fuel rod pressure, annular fuel pellets, and/or axial blankets) on the generation of fission products and the transport of fission products released during LOCA and non-LOCA design basis accidents. These considerations need to be addressed in order for the staff to make a finding that the use of the proposed fuel will not impact assumptions used in the current licensing basis analysis for demonstrating compliance with 10 CFR Part 100, or 10 CFR 50.67, as applicable. For example:

Dose Question 1:

Does the addition of the zirconium diboride coating alter the source term assumptions provided in NUREG-1465 with regard to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release? In particular, does the added zirconium diborate upset current conclusions regarding the radiochemistry of iodines (e.g., does the increased mass of Zr shift the chemical equilibrium of reactions that yield CsI as opposed to ZrI)

Dose Response 1:

The application of ZrB_2 to the CE fleet is basically the same as that implemented for the Westinghouse fleet. The Vantage-5 Fuel Assembly design topical report WCAP-10444, Addendum 1, was reviewed and accepted by NRC on March 12, 1986; where Addendum 1 reported information relative to the use of ZrB_2 in the fuel rod design. Westinghouse has subsequently accrued more than 15 years of satisfactory performance of ZrB_2 design. Based on this experience base and the research and testing that preceded its application in operating plants, [

]^{a,c} Thus, the use of the ZrB_2 fuel will not impact assumptions used in the current licensing basis analysis for demonstrating compliance with 10 CFR Part 100, or 10 CFR 50.67, as applicable.

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]^{a,c} [

][a,c]

[3] WCAP-12921 "Chalk River Irradiation Test of Enriched ZRBs Coatings", February 1991, (Westinghouse Proprietary Class 2)
[4] NUREG-1465

Dose Question 2:

Does the increased helium gas pressure in the fuel pins invalidate assumptions regarding fuel handling accident fission product release iodine scavenging in the spent fuel pool or reactor cavity. Note that the pool decontamination factors in Safety Guide 25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," are predicated on a maximum fuel rod pressurization of 1200 psig. (The assumptions in this safety guide were largely developed from the results of experiments performed by Westinghouse as reported in WCAP-7518-L.)

Dose Response 2:

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Safety Guide 25 identifies a rod internal pressure of 1200 psig as being associated with the determination of the pool scrubbing DF provided by the pool of water and states that the DF will be lower for fuel rod pressures greater than 1200 psig. The Safety Guide also states that with pressures >1200 psig the DF is to be calculated on an individual basis using assumptions comparable in conservatism to those used in the Safety Guide. [

]^{a,c}

Dose Question 3:

Is the fission product migration in an annular pellet different from that of a solid pellet in a manner that would affect previous analysis assumptions related to the fraction of core inventory in the fuel gap, or on the timing of fission product releases following a transient? Is the correlation between burnup and fission product diffusion different for an annular pellet than a solid pellet?

Dose Response 3:

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]^{a,c}

These three questions are offered only as examples. The staff expects Westinghouse to provide a full evaluation of the impact of zirconium diboride on the generation and transport of fission products as currently analyzed in DBA radiological consequence analyses.

Appendix C

Nuclear Regulatory Commission

Round #2 Request for Additional Information

dated October 7, 2003

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NRC Round #2 Request for Additional Information on WCAP-16072-P, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"

1. Where in CEN-372-P-A does it justify fuel clad burst as an acceptable mechanism to prevent DNB Propagation (i.e., excessive clad ballooning)?

2. The DNB Propagation methodology assumes a constant rod internal pressure (i.e. no credit for an increase in rod internal volume accompanying clad ballooning). While this practice is conservative for calculating the amount of clad strain, would it be non-conservative for estimating clad burst (based on an overly conservative calculation of hoop stress) which is being credited to preclude DNB Propagation?

3. Rod internal pressure increases at a faster rate in the ZrB_2 IFBA fuel design than in standard UO_2 rods (due to rapid buildup of Helium). As a result, the fuel rod internal pressure may reach RCS pressure at a lower burnup in a greater number of rods. These higher power fuel rods (with internal pressures exceeding RCS pressure) are more likely to experience DNB during a Non-LOCA event (e.g. SS/SR or IOSGADV+LOP). W-CE methodology credits rod balloon/burst as part of the DNB Propagation Analysis.

   a. During a postulated transient for reload cores containing ZrB_2 IFBA fuel, would a larger number of fuel rods experience balloon/burst for the same number of calculated DNBR failures?

   b. Would it be possible to demonstrate that no fuel pins with rod internal pressure exceeding system pressure will experience DNB during Condition III and Condition IV events?

4. During normal operation, fuel pellets experience cracking and relocation and develop a "rim" consisting of fission products. In addition, dimensional changes in the fuel pellet and clad will lead to pellet-to-clad contact. During postulated transient events which experience DNB, the elevated clad temperature may promote outward creep in fuel rods with internal pressure in excess of system pressure. These same fuel rods may also burst as a result of these elevated temperatures (in the presence of high rod internal pressure).

   a. As the clad creeps away from the fuel pellet and widens the gap, fuel pellet temperature will increase. Has the potential for an increase in fission gas release been evaluated?

   b. The outward creep of the clad will disturb the rim region of the pellet which has partially bonded with the inside surface of the cladding. Has the potential for an increase in radiological source term been evaluated?

   c. Has the potential for fuel fragment relocation in the balloon/burst region been evaluated for both a core cool able geometry and radiological source term perspective?

5. The depletion of B^{10} in the ZrB_2 coating leads to the production of helium and lithium. This topical discusses the impact of helium production, but is silent on any potential impacts of lithium production.

   a. Please describe the evolution of the lithium and any changes in its chemical state with burnup and temperature. In other words, does the lithium react or bond with any of the fission products and/or cladding?

   b. During power maneuvers or anticipated operational occurrences, will the chemical state of the lithium change and/or is it possible for the lithium to reach a gaseous state (and contribute to the rod internal pressure)?

Appendix D

Westinghouse Electric Company LLC Response to

Nuclear Regulatory Commission

Round #2 Request for Additional Information

LTR-NRC-03-56, dated November 3, 2003

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Westinghouse

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Project No.: 700
Our ref: LTR-NRC-03-56

November 3, 2003

Subject: **Response to NRC Round #2 Request for Additional Information Regarding WCAP-16072-P & WCAP-16072-NP, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs" (Proprietary / Non-proprietary)**

References:
1. Fax, B. J. Benney (NRC) to R. Sisk (Westinghouse), "WCAP-16072 Formal RAIs," October 17, 2003.
2. WCAP-16072-P & -NP, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," April 2003.
3. Letter, C. M. Molnar, Westinghouse to USNRC Document Control Desk, "Submittal of WCAP-16072-P, Revision 0, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," (Proprietary/Non-proprietary), LTR-NRC-03-14, April 25, 2003.

Enclosed are copies of Westinghouse Electric Company LLC (Westinghouse) responses to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI), Reference 1, regarding WCAP-16072-P & -NP, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," Reference 2. This topical report was submitted for NRC review and approval on April 25, 2003, Reference 3.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-03-1730 with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit, AW-03-1730.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-03-1730 and should be addressed to B. F. Maurer, Acting Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Copy to: F. M. Akstulewicz, NRC (w/o enclosures)
B. J. Benney, NRC (w/ 3 proprietary & 1 non-proprietary copies)
P. Clifford, NRC (w/o enclosures)

Response to NRC Round #2 Request for Additional Information Regarding WCAP-16072-P & WCAP-16072-NP "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs"

Round #2 RAI #1:

Where in CEN-372-P-A does it justify fuel clad burst as an acceptable mechanism to prevent DNB Propagation (i.e. excessive clad ballooning)?

Response:

The fuel clad high temperature creep and ballooning model is presented in the Request for Additional Information (RAI) on CEN-372-P. The RAI responses are included as Appendix A and the ballooning and rupture models are discussed in response to RAI #3 of CEN-372-P-A. Clad burst was not stated explicitly in RAI #3 as a mechanism to terminate strain and, therefore, prevent DNB propagation. This phenomenon was, however, explicitly identified in a follow-on report for ZIRLO™, CENPD-404-P-A. The burst stress for Zircaloy-4 is an explicit correlation in CEN-372-P-A, as is the high temperature creep equation. [

]^{a,c} Thus, creep was interpreted to be terminated, further strain would not occur, and DNB would not propagate. However, cladding burst as a mechanism to terminate ballooning and propagation has not been applied for Zircaloy-4. [

]^{a,c}

Page 10 of the SER for CEN-372-P-A provides a good discussion and perspective concerning the expected ballooning and rupture behavior of fuel rods in DNB and the impact on adjacent fuel rods. The likelihood of DNB propagation is recognized to be at most confined to a single adjacent fuel rod and no further because of the non-uniform circumferential temperature and non-uniform deformation. Only the surface of the adjacent rod which locally might now experience DNB will in turn deform back into the direction of the original DNB rod. Furthermore, the models for DNB propagation are also agreed to be conservative in the SER. Westinghouse concurs with this SER assessment and further concludes that DNB propagation, while it can be postulated as a phenomenon, is not a likely mechanism for additional fuel rod failure.

Round #2 RAI #2:

The DNB Propagation methodology assumes a constant rod internal pressure (i.e. no credit for an increase in rod internal volume accompanying clad ballooning). While this practice is conservative for calculating the amount of clad strain, would it be non-conservative for estimating clad burst (based on an overly conservative calculation of hoop stress) which is being credited to preclude DNB Propagation?

Response:

The clad burst stress is a function of temperature and can be compared to the high temperature creep rate versus clad stress and temperature. [

]^{a,c} Thus, while it appears that the methodology may be non-conservative, accounting for a pressure reduction due to internal volume increases would result in lower pressure and stress and insufficient strain to cause DNB propagation.

As discussed in Response 1, the [

]^{a,c} However, it is noted that bursting remains a justifiable mechanism for ballooning termination in Zircaloy-4 cladding.

Round #2 RAI #3:

Rod internal pressure increases at a faster rate in the ZrB2 IFBA fuel design than in standard UO2 rods (due to rapid buildup of Helium). As a result, fuel rod internal pressure may reach RCS pressure at a lower burnup in a greater number of rods. These higher power fuel rods (with internal pressures exceeding RCS pressure) are more likely to experience DNB during a Non-LOCA event (e.g. SS/SR or IOSGADV+LOP). W-CE methodology credits rod balloon/burst as part of the DNB Propagation Analysis.

c) During a postulated transient for reload cores containing ZrB2 IFBA fuel, would a larger number of fuel rods experience balloon/burst for the same number of calculated DNBR failures?

d) Would it be possible to demonstrate that no fuel pins with rod internal pressure exceeding system pressure will experience DNB during Condition III and Condition IV events.

Response #3a:

The fuel rods predicted to experience DNB would all experience some degree of ballooning if the internal gas pressure is in excess of the external pressure. Experiencing significant ballooning and/or burst is unlikely because of the conservatism of the maximum pressure calculation and conservatism of the ballooning and burst models. A number of fuel rods experiencing DNB are expected to be in the same region of the core, within an assembly, for example. All these rods are presumed to be failed (i.e., ruptured). Based on the calculated DNB propagation behavior, no additional rods are likely to fail. Thus, the number of failed rods would be the same as the number of initial DNBR failures.

Response #3b:

Westinghouse does not expect it to be possible to demonstrate that no fuel rods would experience DNBR during all Condition III and IV events. For example, Control Rod Ejection may result in DNB. Portions of the fuel rods inherently experience DNB conditions during a LOCA. Criteria more appropriate to predict failed fuel are applied under these accident conditions.

Round #2 RAI #4:

During normal operation, fuel pellets experience cracking and relocation and develop a "rim" consisting of fission products. In addition, dimensional changes in the fuel pellet and clad will lead to pellet-to-clad contact. During postulated transient events which experience DNB, the elevated clad temperature may promote outward creep in fuel rods with internal pressure in excess of system pressure. These same fuel rods may also burst as a result of these elevated temperatures (in the presence of high rod internal pressure).

d) As the clad creeps away from the fuel pellet and widens the gap, fuel pellet temperature will increase. Has the potential for an increase in fission gas release been evaluated?

e) The outward creep of the clad will disturb the rim region of the pellet which has partially bonded with the inside surface of the cladding. Has the potential for an increase in radiological source term been evaluated?

f) Has the potential for fuel fragment relocation in the balloon/burst region been evaluated for both a core coolable geometry and radiological source term perspective?

Response #4a:

The potential for increase in fission gas release has been evaluated in support of CEN-372-P-A. [

]a,c Thus, potential for DNB propagation is concluded to be unlikely.

Response #4b:

The topical report WCAP-16072-P pertains to ZrB_2 coated pellets. It is unlikely that these pellets will bond with the cladding to any greater degree than uncoated pellets. The formation of a pellet rim region only begins at relatively high burnup, i.e., at pellet average burnups of 45-55 MWd/kgU. Furthermore, high burnup fuel rods are depleted and exhibit low power levels consistent with significant reductions in reactivity. It is unlikely that such rods will experience DNB and thus contribute to the calculated doses. Furthermore, the fission gas source term in the low power/high burnup rod is significantly smaller than that assumed for the limiting high power rods in the safety analysis. Thus the potential for a significant increase in radiological source term is negligible.

Response #4c:

Westinghouse has not evaluated the potential for fuel fragmentation and impact on coolable geometry based on postulated rim effects. Neither has the potential for increases in the radiological source terms been evaluated. Tests for which Westinghouse is aware do not indicate significant fuel fragmentation and relocation. Thus, until such data indicate a need, treatment of this phenomenon will not be considered necessary. Existing criteria are considered sufficient.

Round #2 RAI #5:

The depletion of B^{10} in the ZrB2 coating leads to the production of helium and lithium. This topical discusses the impact of helium production, but is silent on any potential impacts of lithium production.

c) Please describe the evolution of the lithium and any changes in its chemical state with burnup and temperature. In other words, does the lithium react or bond with any of the fission products and/or cladding?

d) During power maneuvers or anticipated operational occurrences, will the chemical state of the lithium change and/or is it possible for the lithium to reach a gaseous state (and contribute to the rod internal pressure)?

Response #5a:

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<u>Response #5b:</u>

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]a,c

Appendix E

Nuclear Regulatory Commission

Round #3 Request for Additional Information

dated November 18, 2003

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NRC Round #3 Request for Additional Information on WCAP-16072-P, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"

1. In Section 2.3.1, the topical report states, "The surface roughness of the ZrB_2 IFBA coated fuel pellet surface would be expected to be less than the original uncoated UO_2 fuel pellet surface. However, it is assumed to be the same as UO_2. The coated surface is observed by Westinghouse to be no rougher and no more rigid than that of UO_2." Please demonstrate that the ZrB_2 IFBA coated fuel pellet surface roughness is and will remain less than the uncoated UO_2 pellet.

2. In Section 4.2.3, Large Break Loss-of-Coolant (LBLOCA) and Small Break Loss-of- Coolant (SBLOCA) demonstration analyses reveal that aspects of ZrB_2 IFBA fuel designs, especially impacts of rod internal pressure, have the potential to produce significant changes in the calculated results. The topical states, "Implementation analyses are performed to determine the plant-specific impact of the ZrB_2 IFBA fuel." Cycle-specific evaluations may also be required to ensure that cycle-specific IFBA fuel designs do not invalidate bounding ECCS Performance Analyses. Please describe how licensees implementing this topical report will demonstrate compliance to the acceptance criteria in 10 CFR 50.46(b) and to the reporting criteria in 10 CFR 50.46(a)(3)(i) and (ii).

3. In RAI #11, the staff had concerns that current plant operations (staff training and operating procedures) as well as the MTC surveillance test may be inadequate for an increasing trend in critical soluble boron concentration (and MTC). In response, Westinghouse stated that this "is a plant specific implementation issue". Further, Westinghouse stated it will recommend that procedures be implemented to confirm that the MTC is within its limits at the highest RCS boron concentration expected during the cycle. Please describe the recommended actions licensees must complete prior to implementing this topical report.

4. In response to RAI #8, Westinghouse stated that their evaluation of cladding collapse in the plenum region of the rods demonstrated that cladding collapse would not occur if the radial support offered to the cladding by the plenum spring was factored into the calculation. The staff does not approve of this deviation from established methodology. Please provide analyses demonstrating fuel rod failure will not occur due to clad flattening (collapse). No credit may be taken for any radial support provided by the plenum spring.

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Appendix F

Westinghouse Electric Company LLC Response to

Nuclear Regulatory Commission

Round #3 Request for Additional Information

LTR-NRC-03-70, dated December 5, 2003

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Westinghouse

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Project No.: 700
Our ref: LTR-NRC-03-70

December 5, 2003

Subject: Response to Round #3 Request for Additional Information Regarding WCAP-16072-P & -NP, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs " (Proprietary / Non-proprietary)

References:
1. E-Mail form B. Benney (NRC) to R. Sisk (Westinghouse), Round 3 RAIs on ZrB_2 Implementation Topical Report, 11/18/2003.
2. WCAP-16072-P & -NP, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs", April 2003
3. Letter, H. A. Sepp (W) to USNRC Document Control Desk, "Submittal of WCAP-1602-P, Revision 0, Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs, (Proprietary/Non-proprietary)", LTR-NRC-03-14, April 25, 2003

Enclosed are copies of Westinghouse Electric Company LLC (Westinghouse) responses to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI), Reference 1, regarding WCAP-16072-P & -NP, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs", Reference 2. This topical report was submitted for NRC review and approval on April 25, 2003, Reference 3.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-03-1745 with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit, AW-03-1745.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR 9.17(a)(4). Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

This material is for your internal use only and may be used solely for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Office of Nuclear Reactor Regulation without the expressed prior written approval of Westinghouse.

Correspondence with respect to any Application for Withholding should reference AW-03-1745 and should be addressed to Ian C. Rickard, Program Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company.

Very truly yours,

I. C. Rickard

I. C. Rickard, Program Manager
Regulatory Compliance and Plant Licensing

Copy to: F. M. Akstulewicz, NRC (w/o enclosures)
B. J. Benney, NRC (w/ 3 proprietary & 1 non-proprietary copies)
P. Clifford, NRC (w/o enclosures)

Enclosure

Non-Proprietary Response to NRC Round #3 Request for Additional Information

WCAP-16072-P & -NP

"Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs"

Round #3 RAI #1:

In Section 2.3.1, the topical report states, "The surface roughness of the ZrB_2 IFBA coated fuel pellet surface would be expected to be less than the original uncoated UO_2 fuel pellet surface. However, it is assumed to be the same as UO_2. The coated surface is observed by Westinghouse to be no rougher and no more rigid than that of UO_2". Please demonstrate that the ZrB_2 IFBA coated fuel pellet surface roughness is and will remain less than the uncoated UO_2 pellet.

Response:

Surface roughness measurements of the uncoated, standard UO_2 pellets and the coated UO_2 pellets for fuel pellets fabricated at about the same time, recently as well as several years ago, were reviewed. [

]^{a,c} The observation that the coated surface is no rougher than the uncoated UO_2 surface is correct. Furthermore, the stated assumption that the coated pellet surface roughness is the same as the uncoated pellets is an appropriate assumption.

Westinghouse currently applies a surface roughness specification for the fabrication of standard UO_2 pellets that is identical to the specification for ZrB_2 coated pellets. Fabrication process control procedures are in place to ensure the specifications are met. [

]^{a,c} Westinghouse will continue to impose a fabrication surface roughness specification for ZrB_2 pellets that will be assured through appropriate fabrication quality assurance procedures and appropriately accounted for in the design and safety analyses

Round #3 RAI #2:

In Section 4.2.3, Large Break Loss-of-Coolant (LBLOCA) and Small Break Loss-of-Coolant (SBLOCA) demonstration analyses reveal that aspects of ZrB_2 IFBA fuel designs, especially impacts of rod internal pressure, have the potential to produce significant changes in the calculated results. The topical states, AImplementation analyses are performed to determine the plant-specific impact of the ZrB_2 IFBA fuel@. Cycle-specific evaluations may also be required to ensure that cycle-specific IFBA fuel designs do not invalidate bounding ECCS Performance Analyses. Please describe how licensees implementing this topical report will demonstrate compliance to the acceptance criteria in 10 CFR 50.46 (b) and to the reporting criteria in 10 CFR 50.46 (a)(3)(i) and (ii).

Response:

The quoted statement, which appears in the topical report in both the LBLOCA (Section 4.2.3.1) and SBLOCA (Section 4.2.3.2) discussions is, in a sense, redundant, being simply a statement of normal process. The statement is not setting a new process in place, rather, it is merely a reflection of how the normal reload analysis proceeds, regardless of the presence of new fuel features or methodology changes.

Specifically, the first step in the reload analysis process is to determine the need for the performance of full scope analyses (in this case LOCAs). This determination is accomplished by reviewing key parameter checklists for LOCA analyses to determine if anything associated with the upcoming reload cycle exceeds (in a non-conservative direction) specific parameter values associated with the bounding analysis results for the existing Analysis of Record (AOR). If no key parameter values are exceeded, a new analysis to replace the existing AOR is not required and the reload can be processed via 10 CFR 50.59. However, where a key parameter value would result in the AOR no longer remaining bounding, full scope LOCA analyses are performed to assure that all 10 CFR 50.46(b) acceptance criteria are satisfied.

When new fuel design feature(s) or analysis methods are implemented, a similar process is followed. That is, new analyses are performed to assess the impact of new fuel design feature(s) and/or methods relative to the existing bounding AOR results to assure that all 10 CFR 50.46(b) acceptance criteria are satisfied. If the results of these analyses are more limiting than the existing bounding AOR results but do not require any changes to regulatory limits (e.g., Technical Specification Limiting Conditions of Operation), the utility updates the applicable UFSAR sections pursuant to 10 CFR50.71 and 10 CFR 50.59. If the new results require changes to Technical Specification Limiting Conditions of Operation, the utility submits the appropriate license amendment pursuant to 10 CFR 50.90.

With respect to the reporting requirements of 10 CFR 50.46(a)(3)(i) and (ii), the effect of changes is tracked on a utility specific basis. When changes do not exceed the regulatory significance threshold of 50°F, the effect of change(s) is submitted to the NRC by the licensee in an annual 10 CFR 50.46 report. If changes exceed the regulatory significance threshold of 50°F, the effect of change(s) is submitted to the NRC by the licensee within the 30-day time period specified in the regulation. In general, such a report would document the effect and that it resulted from the implementation of some new fuel design feature or methodology change which had already been reviewed and approved by the NRC, as for example in this case the implementation of ZrB_2. This is exactly the process that was followed when the Westinghouse-developed ZIRLO™ cladding material was implemented for CE fuel designs. A topical report

(CENPD-404-P-A, Rev. 1) was submitted, reviewed and approved. Utilities updated their COLR references, as necessary, and the reload analysis described above was employed on the first introduction of a batch of ZIRLO™ clad fuel. Because methodology changes other than those strictly associated with implementation of ZIRLO™ were performed, it was determined that the change in Peak Clad Temperature (PCT) exceeded the regulatory significance threshold of 50°F. Consequently, licensees submitted a 30-day letter acknowledging the significance of the change and that it was due to the implementation of a previously NRC approved fuel design change and previously approved methodology change. No further action was required.

Round #3 RAI #3:

In RAI #11, the staff had concerns that current plant operations (staff training and operating procedures) as well as the MTC surveillance test may be inadequate for an increasing trend in critical soluble boron concentration (and MTC). In response, Westinghouse stated that this "is a plant specific implementation issue". Further, Westinghouse stated it will recommend that procedures be implemented to confirm that the MTC is within its limits at the highest RCS boron concentration expected during the cycle. Please describe the recommended actions licensees must complete prior to implementing this topical report.

Response #3:

For some plants containing ZrB_2 IFBA, the most positive MTC may occur not at beginning of cycle (BOC) but at some later time within the first third of the cycle where the critical boron concentration (CBC) is at its greatest value. For CE plants using ZrB_2 IFBA, the difference between the most positive MTC and the BOC value is expected to be small (<1 pcm/°F). This difference is well within the MTC uncertainty allowance used in the core design and safety analysis processes. Current Westinghouse core design procedures already require that the maximum value of the predicted MTC be confirmed to be within its limit at all burnups during the cycle. In addition to this existing requirement, Westinghouse will recommend that the MTC surveillance requirement be modified to indicate that if the cycle maximum HFP CBC is more than 100 ppm greater than the BOC HFP value and if the results of the BOC MTC tests indicate a difference between prediction and measurement that is larger than the design tolerance (1.6 pcm/°F) then an additional MTC surveillance should be performed at HFP within 7 EFPD of the core burnup corresponding to the cycle maximum predicted HFP CBC. Note that this approach is consistent with the method approved by the NRC in CE-NPSD-911 to eliminate the End-of-Cycle MTC surveillance requirement.

Round #3 RAI #4:

In response to RAI #8, Westinghouse stated that their evaluation of cladding collapse in the plenum region of the rods demonstrated that cladding collapse would not occur if the radial support offered to the cladding by the plenum spring was factored into the calculation. The staff does not approve of this deviation from established methodology. Please provide analyses demonstrating fuel rod failure will not occur due to clad flattening (collapse). No credit may be taken for any radial support provided by the plenum spring.

Response #4:

The cladding creep behavior during creep ovalization in the plenum region is the same as creep ovalization in the fuel region. Support from the plenum spring coil is similar to support from the fuel pellets for the case of finite axial pellet-to-pellet gaps. Support from the fuel pellet is a well established and accepted methodology. Thus, Westinghouse concludes that radial support from the plenum spring coils is not a significant deviation from established methodology.

The plenum spring material is stainless steel. Creep of the stainless steel in the plenum region is insignificant. Lateral load carrying capability for the typical plenum spring captured within the cladding tube is significant. [

]^{a,b,c} Thus, the data demonstrate that collapse in the plenum region cannot occur if a standard plenum spring is present.

Section 4.2.2.4 of WCAP-16072-P states that evaluations of cladding collapse in the plenum region would not occur if the radial support of the plenum spring is taken into account. NRC RAI Round 1 Response #8 states that credit for the radial support from the plenum spring was necessary because of the addition of lower initial gas pressure combined with conservative values for other input parameters in the cladding collapse analysis. Response #8 states that future cladding collapse analyses may or may not utilize this credit. Plenum collapse evaluations are expected to be performed using the previously approved CEPANFL for Westinghouse CE plants. However, performing less conservative plenum collapse analyses is not justified in view of the insignificant probability that plenum collapse can occur. Westinghouse concludes that credit for plenum spring support, which has been [ ]^{b}, and which is quite similar to established methodology for pellet support, is appropriate.

Appendix G

Nuclear Regulatory Commission

Round #4 Request for Additional Information

dated January 20, 2004

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NRC Round #4 Request for Additional Information on WCAP-16072-P, "Implementation of ZrB_2 Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs"

1. ZrB_2 IFBA designs were first defined in WCAP-10444-P-A (1986) as part of the VANTAGE5 fuel design. The build-up of rod internal pressures exhibited by the VANTAGE5 fuel rod designs (and later by the VANTAGE+ fuel rod design) do not appear as dramatic as the CE fuel rod designs.

   a. Describe any differences in fuel rod design and/or ZrB_2 loading which would promote the differences in rod internal pressure between Westinghouse and CE fuel rods.

   b. Describe any differences in fuel performance methodology which would promote the differences in rod internal pressure between Westinghouse and CE fuel rods.

   c. Employing one of the approved fuel performance methodologies consistently applied to a sample Westinghouse and CE fuel assembly design, quantify the differences in rod internal pressures a function of burnup.

   d. Employing a best-estimate approach (e.g., an actual single rod power history) consistently applied to a sample Westinghouse and CE fuel assembly design, quantify the differences in rod internal pressure as a function of burnup.

2. Describe the build-up of rod internal pressures in ZrB_2, Erbia, and UO_2 fuel rods for the CE fuel designs. Include a discussion of the relative likelihood of experiencing DNB (during a Condition III and IV event) with rod internal pressure greater than system pressure. A core wide pin census of rod power and rod internal pressure as a function of burnup would be helpful to comprehend the impact (e.g., number of fuel rods) for the different fuel types.

3. The current CE mechanistic approach to DNB Propagation is defined in CEN-372-P-A (1990). Approval of this document established a strain limit of [ ]^{a,c} with no provisions for rod burst. Documentation supporting rod burst (as a mechanism credited to halt rod strain) was first introduced within CENPD-404-P-A (2001) for ZIRLO application in CE fuel assembly designs and later in Addendum 1 to WCAP-8963-P-A (review pending).

   a. Is the [ ]^{a,c} strain limit still valid or will fuel clad burst prior to ever achieving this degree of ballooning (all fuel clad materials)?

   b. Using burst stress criteria for CE Zr-4, demonstrate that past and present CE core reloads containing CE Zr-4 fuel rods (other than ZIRLO) would not experience rod burst during Condition III and IV events.

   c. Specifically identify which events would challenge strain limits and which events would more likely challenge clad burst limits.

   d. How will the application of the CE mechanistic approach (Addendum 1 to WCAP-8963-P-A) change the Westinghouse fuel rod design and/or fuel management guidelines.

4. Currently licensed fuel mechanical design criteria include [

.]^{a,c} Please address this apparent conflict.

5. An allowable rod strain/burst approach introduces a new fuel rod failure mechanism for Condition III and IV events.

   a. Provide evidence that the risks associated with this failure mechanism have been adequately communicated to the licensees.

   b. Demonstrate that the past selection of LPD and DNB limiting Condition III and IV events remains limiting for this new fuel rod failure mechanism. For example, would a PPCS malfunction with LOP become more limiting with respect to rod strain/burst than the currently analyzed IOSGADV + LOP event?

   c. Demonstrate that fuel clad integrity is maintained in fuel rods with high internal rod pressures during a rapid depressurization events which may not violate current SAFDLs. For example, would a SGTR or MSLB combined with a new single active failure or operator error (selected to further challenge rod strain/burst) yield a breach in clad integrity?

   d. The burst geometry will be influenced by internal-to-external differential pressure, clad material, creep properties, irradiation hardening, clad temperature, etc. Many variables also exist on the stability of the fuel pellet during balloon/burst and its potential impact on radiological source term, fuel fragmentation/relocation, and coolable geometry. Allowing rod burst created uncertainty due to these variables. Furthermore, the results of ongoing research may generate real safety concerns for operating plants which credit rod burst. Please address these concerns.

6. The SER for CEN-372-P-A states that "...radially-oriented hydrides have been shown to have a detrimental effect on cladding strength and ductility under tensile conditions." Furthermore, the SER states, "It has been demonstrated that these hydrides will precipitate in the circumferential direction as long as compressive cladding stresses are maintained, but will precipitate in the undesirable radial direction if cladding tensile stresses are maintained above a given stress range during cool down.

   a. To preclude radially-oriented hydride precipitates, demonstrate that the tensile stress on the cladding remains below 10 ksi during a plant shutdown or anticipated operational occurrence.

   b. For all fuel designs, demonstrate that the maximum tensile hoop stress experienced during the limiting transient remains sufficiently low to preclude hydride reorientation.

Appendix H

Westinghouse Electric Company LLC Response to

Nuclear Regulatory Commission

Round #4 Request for Additional Information

LTR-NRC-04-10, dated February 3, 2004

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Westinghouse

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Our ref: LTR-NRC-04-10

February 3, 2004

Enclosed is:

1. One (1) copy of "Responses to RAIs on WCAP-16072-P and Addendum 1 to WCAP-8963-P-A (Proprietary)

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-04-1790 (Nonproprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Nonproprietary).

This information is being submitted by Westinghouse Electric Company LLC to respond to RAIs faxed from Brian Benney to Rob Sisk on January 20, 2004.

This submittal contains proprietary information of Westinghouse Electric Company LLC. In conformance with the requirements of 10 CFR Section 2.790, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-04-1790 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: D. Holland
B. Benney
E. Peyton

Responses to RAIs on WCAP-16072-P and Addendum 1 to WCAP-8963-P-A

1. ZrB2 IFBA designs were first defined in WCAP-10444-P-A (1986) as part of the VANTAGE5 fuel design. The build-up of rod internal pressures exhibited by the VANTAGE5 fuel rod design (and later by the VANTAGE+ fuel rod design) do not appear as dramatic as the CE fuel rod designs.

    a. Describe any differences in fuel rod design and/or ZrB2 loading which would promote the differences in rod internal pressure between Westinghouse and CE fuel rods.

    b. Describe any differences in fuel performance methodology which would promote the differences in rod internal pressure between Westinghouse and CE fuel rods.

    c. Employing one of the approved fuel performance methodologies consistently applied to a sample Westinghouse and CE fuel assembly design, quantify the differences in rod internal pressure as a function of burnup.

    d. Employing a best-estimate approach (e.g. an actual single rod power history) consistently applied to a sample Westinghouse and CE fuel assembly design, quantify the differences in rod internal pressure as a function of burnup.

Response 1:

Fuel rod internal pressures in Westinghouse ZrB_2 IFBA rods are similar to the CE ZrB_2 IFBA rods. [ ]^{a,c} in the Westinghouse and in the CE fuel rods.

Best-estimate and maximum internal pressures in the typical non-IFBA Westinghouse rod (UO_2) are shown in Figure 5 of Attachment 3 in Section L of WCAP-15063-P-A. Best-estimate end of life pressure is approximately [ ]^{a,c} as shown in Figure 5. The [ ]^{a,c} pressure in a typical UO_2 CE fuel rod, nearly a best-estimate pressure history, is shown in Figure 3.2-3 of CENPD-382-P-A (also shown is an erbia fuel rod). The CE UO_2 rod is approximately [ ]^{a,c} at end of life, similar to the Westinghouse rod. In addition the best-estimate pressure build-up in the Westinghouse design and the CE design are nearly identical. These examples are both peak linear heat generation rate (LHGR) rods, starting at about 10 kw/ft and then followed by a decreased LHGR. Additional details of these examples are available in the references.

The maximum pressure for the ZrB_2 IFBA rods shown in Figures 4.2-3 and 4.2-4 of WCAP-16072-P indicate an increase in pressure early in life that is due to the partial pressure from helium generated from the conversion of B^{10} and released from the IFBA coating. The model for B^{10} depletion and helium release is shown in Figure 3-1 of WCAP-16072-P for the Westinghouse PAD code and for the CE FATES3B code. The helium component generated and released from identical ZrB_2 coating designs is virtually identical in both codes. This component is shown in Figure 14 of Attachment 3 in Section L of WCAP-15063-P-A for Westinghouse fuel. It has the same pressure increase as seen with CE fuel. [

]^{a,c} The release of helium early in life is known to be very conservative. [

]^{a,c} Although not quantified at this time, the actual release is experimentally shown to be slower and will not approach 100% until high burnup is achieved, beyond the first cycle of operation. Consequently, [

]^{a,c} This is especially true since the backfill pressure of the ZrB_2 IFBA rods has been reduced compared to a UO2 rod to accommodate helium release. As stated in WCAP-16072-P, Westinghouse has ongoing programs to determine the helium release behavior of ZrB_2 coatings.

It is concluded that maximum internal pressure differences are not due to the design nor are the differences due to the ZrB_2 IBFA. The maximum pressure differences are due to differences in approved licensing methodology.

Response 1a:

The Westinghouse and CE fuel rods are supported in assembly arrays ranging from the 14x14 CE/Westinghouse assemblies to the 17x17 Westinghouse assemblies. Rod diameters, active fuel lengths, and plenum lengths differ from core to core. Furthermore, the fuel temperatures, fission gas release, and internal pressure of the particular burnable absorber fuel rod can result in differences from core to core and between Westinghouse and CE fuel designs of different types. Requirements for burnable absorber content, including the ZrB_2 loading depends on the core design, operating cycle lengths, and power level. [

]^{a,c} This is demonstrated in Response 1. Westinghouse concludes that this is generally the case and pressure histories may differ because imposed linear heat rates may differ; the hotter the rod the higher the pressure.

Response 1b:

Westinghouse typically applies [ ]^{a,c} uncertainties to Westinghouse plants to obtain the maximum internal hot gas pressure. Consequently, the pressure uncertainty builds up in direct proportion with pressure and burnup. This is clearly demonstrated in Figure 5 of Attachment 3 in Section L of WCAP-15063-P-A. Westinghouse applies [

]^{a,c} This is defined as a limiting radial peaking factor. Management of the fuel in the CE core ensures this limiting radial peaking factor is not violated. It can be seen in Figure 3.2-3 of CENPD-382-P-A [

]^{a,c} Although the pressure has increased early in the ZrB_2 IFBA rods as shown in Figures 4.2-3 and 4.2-4 of WCAP-16072-P, the pressure also levels out at approximately [ ]^{a,c} Because the limiting radial peaking factor

approach results in higher LHGRs than the RMS analyses, the maximum internal gas pressure is generally higher in the CE fuel designs than in the Westinghouse fuel designs.

Response 1c:

It can be concluded from Responses 1, 1a, and 1b that applying one of the methodologies to a Westinghouse fuel rod or to a CE fuel rod will result in comparable results.

Response 1d:

Results of the best-estimate calculations of internal pressures in the Westinghouse design and in the CE design will be comparable as described in Response 1.

2. Describe the build-up of rod internal pressures in ZrB2, Erbia, and UO_2 fuel rods for the CE fuel designs. Include a discussion of the relative likelihood of experiencing DNB (during a Condition III and IV event) with rod internal pressure greater than system pressure. A core wide pin census of rod power and rod internal pressure as a function of burnup would be helpful to comprehend the impact (e.g. number of fuel rods) for the different fuel types.

Response 2:

Build up of rod internal pressure in a ZrB_2 rod, in an erbia rod, and in a UO_2 rod is described in Response 1. As indicated in Response 1, ZrB_2 rods will have higher internal rod pressure earlier in life compared to erbia and UO_2 rods, primarily due to the conservative assumptions on helium release from the ZrB_2 coating. However, NCLO critical pressure will not be violated. The fuel rods which are postulated to be in DNB for Condition III and IV events are accounted for in each plant's dose calculations. The material behavior of such rods in DNB may demonstrate that no DNB propagation will occur, in which case no additional rods will need to be included in the dose calculations for any of these rod types.

3. The current CE mechanistic approach to DNB Propagation is defined in CEN-372-P-A (1990). Approval of this document established a strain limit of [ ]^{a,c} with no provisions for rod burst. Documentation supporting rod burst (as a mechanism credited to halt rod strain) was first introduced within CENPD-404-P-A (2001) for ZIRLO application in CE fuel assembly designs and later in Addendum 1 to WCAP-8963-P-A (review pending).

   a. Is the [ ]^{a,c} strain limit still valid or will fuel clad burst prior to ever achieving this degree of ballooning (all fuel clad materials)?

   b. Using burst stress criteria for CE Zr-4, demonstrate that past and present CE core reloads containing CE Zr-4 fuel rods (other that ZIRLO) would not experience rod burst during Condition III and IV events.

   c. Specifically identify which events would challenge strain limits and which events would more likely challenge clad burst limits.

   d. How will the application of the CE mechanistic approach (Addendum 1 to WCAP-8963-P-A) change the Westinghouse fuel rod design and/or fuel management guidelines.

Response 3:

Approval of CEN-372-P-A established a strain limit [ ]^{a,c} with provisions for rod rupture as evidenced in the SER by the following reference: "CE has used a realistic film boiling coefficient and rupture strain model to predict cladding temperatures and strains for the analysis of DNB propagation for the limiting steam line break and locked rotor accidents for the 14x14 and 16x16 plants.". Thus, the cladding rupture model for Zircaloy-4 was implicitly recognized in CEN-372-P-A. Cladding rupture is not precluded by any designs or evaluations described in CEN-372-P-A. The introduction of new cladding materials will impact the rupture properties, fuel designs will not. The ZIRLO™ rupture model has been explicitly identified in CENPD-404-P-A and remains valid.

Response 3a:

The[ ]^{a,c} strain limit is still valid, and the fuel cladding may rupture prior to ever achieving this degree of ballooning for any clad materials. The specific accident conditions would determine whether the rupture model is violated prior to the [ ]^{a,c} strain limit. This outcome is implied in CEN-372-P-A and further clarified in CENPD-404-P-A and is considered by Westinghouse to be both licensed and acceptable.

Response 3b:

As demonstrated in Responses 1 and 1a, past and present CE core reloads containing CE Zircaloy-4 fuel rods (other than ZIRLO™) do not need to exclude the possibility that the rod rupture mechanism would occur during Condition III and IV events.

Response 3c:

Extremely high cladding stress and temperature DNB transients would typically challenge the rupture mechanism. Long duration, lower stress, and lower temperature DNB

transients would typically challenge the strain limit. The analyses of DNB events, however, incorporate numerous conservatisms which are described in CEN-372-P-A, CENPD-404-P-A, WCAP-16072-P, and Addendum 1 to WCAP-8963-P-A. Most importantly, for example,

1. **DNB is assumed to begin at the 95% probability DNBR SAFDL. DNB does not occur until the DNBR is reduced to 1.0. As a result of this assumption the high clad temperatures are predicted in the DNB propagation mechanistic approach due to the Condie-Bengston IV heat transfer correlation.**
2. **Maximum pressure is a conservative bounding analysis or the upper bound from an RMS analysis.**
3. **Cladding ballooning is assumed to be symmetrical. True ballooning strains are local and would terminate propagation and limit DNB to the single adjacent rod.**
4. **Symmetrical ballooning, if it did occur, would rapidly reduce the internal pressure, terminate further ballooning, and prevent DNB propagation.**
5. **The current ZrB_2 IFBA helium release model results in 100% of the helium to be released in the first cycle of operation. Although the B^{10} depletion is modeled on a best-estimate basis, the release would take longer to occur and would not be expected to be complete until a later cycle and eventually not even reach 100% release.**

Westinghouse concludes that postulated DNB transients may occur but DNB propagation is insignificant at most, and that ballooning and/or rupture is an artifact of the conservative nature of the analysis.

Response 3d:

The application of the CE mechanistic approach (Addendum 1 to WCAP-8963-P-A) will not change the Westinghouse fuel rod design limiting criteria. As is done currently, Westinghouse fuel rod design and/or fuel management guidelines will be maintained which conform to allowable fuel rod design limits.

4. Currently licensed fuel mechanical design criteria include [

]^{a,c} Please address this apparent conflict.

Response 4:

[

]^{a,c}

The 1% strain limit for Condition III transients was a self imposed (for CE plants) mechanical constraint applied in consideration of the potential for PCI failures. The limit only addresses that amount of mechanical strain, independent of the possibility of DNB, induced in the clad as a result of pellet swelling due to thermal expansion caused by power density changes. The [ ]^{a,c} strain limit imposed by DNB propagation concerns is strictly a clad ballooning issue due to the stresses caused by clad temperature changes. The two limits are independent of one another.

[

]^{a,c}

5. An allowable rod strain/burst approach introduces a new fuel rod failure mechanism for Condition III and IV events.

   a. Provide evidence that the risks associated with this failure mechanism have been adequately communicated to the licensees.

   b. Demonstrate that the past selection of LPD and DNB limiting Condition III and IV events remains limiting for this new fuel rod failure mechanism. For example, would a PPCS malfunction with LOP become more limiting with respect to rod strain/burst than the currently analyzed IOSGADV+LOP event.

   c. Demonstrate that fuel clad integrity is maintained in fuel rods with high internal rod pressures during rapid depressurization events which may not violate current SAFDLs. For example, would a SGTR or MSLB combined with a new single active failure or operator error (selected to further challenge rod strain/burst) yield a breach in clad integrity.

   d. The burst geometry will be influenced by internal-to-external differential pressure, clad material, creep properties, irradiation hardening, clad temperatures, etc. Many variables also exist on the stability of the fuel pellet during balloon/burst and its potential impact on radiological source term, fuel fragmentation/relocation, and coolable geometry. Allowing rod burst created uncertainty due to these variables. Furthermore, the results of ongoing research may generate real safety concerns for operating plants which credit rod burst. Please address these concerns.

Response 5:

Westinghouse is not introducing a new fuel rod failure mechanism for Condition III and IV events. Per NUREG-0800 (Standard Review Plan, Section 4.2), fuel rod failures are permitted during postulated accidents. Rod bursting is specifically identified and addressed as a fuel rod failure mechanism in NUREG-0800. The bursting mechanism results from "swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding."

With respect to DNB Condition III and IV transients:

By their nature, the licensed bounding DNB methodologies force the cladding into high stresses and temperatures during a postulated DNB transient. The cladding rupture stress is a fundamental approved material property of the cladding which cannot be ignored (refer to the NRC's MATPRO document for Zircoloy-4 material properties). The thermal rupture stress relationship is licensed both with the Zircaloy-4 and ZIRLO™ material as previously cited. The high temperature cladding creep is also a fundamental approved material property of the cladding which cannot be ignored. The high temperature creep for both Zircaloy-4 and ZIRLO™ is licensed as previously cited.

The cladding rupture mechanism has existed in analysis since both the basic material properties and DNB bounding methodologies have been licensed. Because the DNB transient bounding methodology potentially takes the cladding stress and temperature into the regime of both high-temperature creep, and rupture, neither mechanism can be ignored. However, the possibility of high ballooning strains and/or cladding rupture is generally not expected on a best-estimate basis. Ballooning strains and /or rupture are

primarily an artifact of analyzing DNB conditions on a bounding basis. Therefore the analysis of the potential for rod burst in the DNB mechanistic propagation methodology is appropriate.

With respect to non-DNB Condition III and IV transients:

Response to RAI Number 24 of CENPD-404-P-A documents that a review of the transients in Table 7.3-1 of CENPD-404-P-A showed that the Seized Rotor/Sheared Shaft event (one of the two events identified as having the highest probability of reaching [ ]^{a,c} has a maximum clad temperature during the event of only ~700 °F. Typically during DNB events, the clad temperature remains well below ~700 °F until DNB actually occurs, at which point the clad begins to heat up. Non-DNB events are typically associated with rapid power density changes in which insufficient time is available for clad heat up.

If one considers a postulated event with RCS depressurization where SAFDLs are not violated, the cladding temperature would remain below [ ]^{a,c} Assuming the fuel rods at maximum internal hot gas pressure in combination with a nearly complete RCS depressurization, and a conservatively high maximum clad temperature of [ ]^{a,c} the maximum cladding stress remains below the cladding yield stress and well below the cladding ultimate stress (Section 5 of CENPD-404-P-A). Thus, rupture of the cladding (which would be expected only when the ultimate stress is exceeded), cannot occur in such an event. Further, such a postulated event would not breach the cladding integrity.

New safety concerns which may surface as a result of ongoing research should be addressed at such time as they become evident.

6. The SER for CEN-372-P-A states that "...radially-oriented hydrides have been shown to have a detrimental effect on cladding strength and ductility under tensile conditions". Furthermore, the SER states, "It has been demonstrated that these hydrides will precipitate in the circumferential direction as long as compressive cladding stresses are maintained, but will precipitate in the undesirable radial direction if cladding tensile stresses are maintained above a given stress range during cool down".

   a. To preclude radially-oriented hydride precipitates, demonstrate that the tensile stress on the cladding remains below 10 ksi during a plant shutdown or anticipated operational occurrence.

   b. For all fuel designs, demonstrate that the maximum tensile hoop stress experienced during the limiting transient remains sufficiently low to preclude hydride reorientation.

Response 6:

The SER for CEN-372-P-A states that CE design methodology shows that the maximum cladding tensile hoop stress that could be experienced by CE fuel designs at the critical pressure limit during normal and anticipated operational occurrences will be less than that required to initiate adverse hydride reorientation in the radial direction. The critical pressure criterion has not changed, and will continue to be applied. It is the intention to address adverse hydride reorientation for conditions where the plant will recover and restart (Condition I & II). It is not intended to address reorientation for events where restart is not possible without further evaluation of fuel system damage (Condition III & IV events). CE designs for Zircaloy-4 (including ZrB_2 IFBA) will continue to conform to the license requirements in CEN-372-P-A.

With respect to ZIRLO™, CENPD-404-P-A states that the tensile stresses and peak temperatures for operation at NCLO conditions were concluded to be well below the magnitudes that might result in adverse hydride reorientation. Therefore, operation with ZIRLO™ will be similar to operation with Zircaloy-4. Consequently, the potential for stress induced hydride reorientation is not affected by operation of fuel rod internal pressures limited by NCLO for ZIRLO™. The critical pressure criterion has not changed, and will continue to be applied. CE designs for ZIRLO™ (including ZrB_2 IFBA) will continue to conform to the license requirements in CENPD-404-P-A.

To initiate hydride re-orientation, the cladding temperature must first increase substantially in a transient to cause the precipitated hydrides to go back into solution and then reorient with re-precipitation in the adverse direction during cool down. Maximum cladding temperature was estimated for CEN-372-P-A based on the assumption that the linear heat generation rate (LHGR) approached the centerline melt SAFDL (RAI Number 4 of CEN-372-P-A). It was concluded that this temperature level would not result is sufficient re-solution of hydrides to have a significant impact on re-orientation and loss of cladding ductility. In addition, as stated in Section 3.3.3 of CEN-372-P-A, the maximum tensile stress during moderate frequency events [

]^{a,c} **Maximum stress is based on the low RCS pressure trip because as soon as the reactor trips, the power level, internal pressure, and clad stress will rapidly decrease from the maximum stress level.**

Thus, adverse hydride re-orientation is precluded in normal operation and anticipated operational occurrences.

ATTACHMENT

The referenced figures in the RAI Responses are attached in the following order:

1. **WCAP-15063-P-A, Figure 5 of Attachment 3 in Section L**
2. **CENPD-382-P-A, Figure 3.2.3.**
3. **WCAP-16072-P, Figure 4.2-3 and Figure 4.2-4.**
4. **WCAP-16072-P, Figure 3-1.**
5. **WCAP-15063-P-A, Figure 14 of Attachment 3 in Section L**

a, c

Figure 5
PAD 4.0 Typical Rod Internal Pressure Versus Rod Average Burnup (MWD/MTU)

Fig. 3.2-3. Maximum Internal Gas Pressure

16x16 Fuel Design

a,c

Rod Internal Pressure, PSIA
(Thousands)

Figure 4.2-3 Maximum Internal Gas Pressure
14x14 Fuel Design

a, c

Figure 4.2-4 Maximum Internal Gas Pressure
16x16 Fuel Design

a, c

Figure 3-1

PAD/FATES3B Comparison of IFBA Model

a, c

a,c

WCAP-16072-NP-A
Westinghouse Non-Proprietary Class 3

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